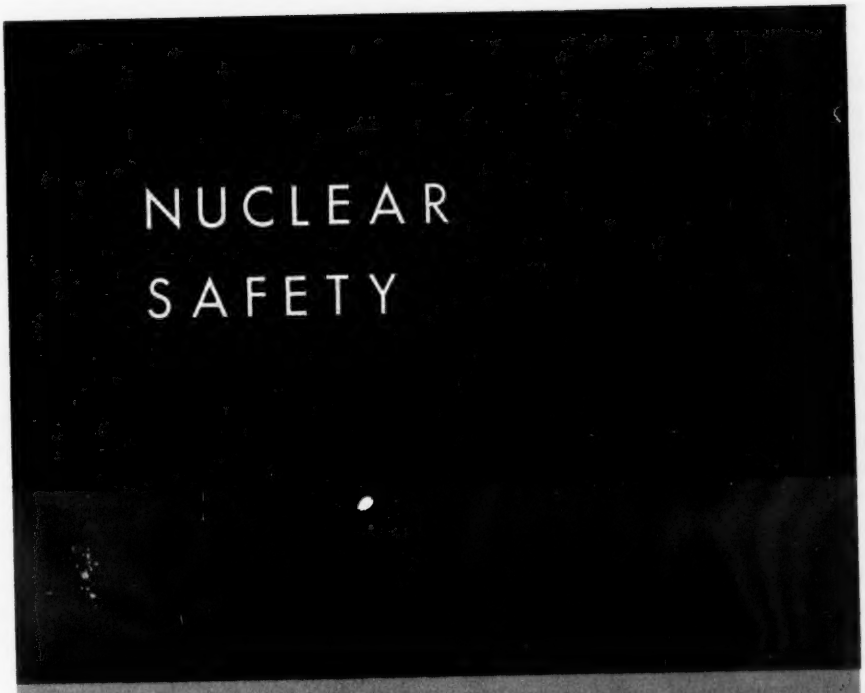
NUCLEAR SAFETY



Prepared for U. S. ATOMIC ENERGY COMMISSION
by OAK RIDGE NATIONAL LABORATORY

● JUNE 1960

● VOLUME 1

● NUMBER 4

foreword

This quarterly journal is a Technical Progress Review prepared by the Oak Ridge National Laboratory at the request of the Office of Technical Information, U. S. Atomic Energy Commission. This Review is intended to assist those interested in keeping abreast of significant developments in the field of nuclear safety. *Nuclear Safety* is not a comprehensive abstract of all literature published in this field during a given quarter; rather it is a mechanism for presenting concise reviews of selected subjects as prevailing interest and available information warrant.

Coverage of the Review is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission. Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, and related operations are also treated. Safety in the use of radioisotopes in industry, medicine, and research is excluded, as are most topics considered to be in the province of health physics. Even with these exclusions, nuclear safety overlaps such diverse fields as nuclear physics, solid-state physics, mechanics, chemistry, meteorology, geology, seismology, metallurgy, law, and nearly all branches of engineering. The authors will therefore review material from these fields which, in their opinion, has a direct bearing on nuclear safety.

Three distinctly different types of articles are in this issue of *Nuclear Safety:* reviews of current literature, special review articles on specific topics, and comparative studies of various aspects of U. S. power reactors. The editors feel that each of these articles makes a necessary and distinctive contribution to this journal. The special review articles permit discussion of pertinent subjects which cannot be adequately considered by reference to only the current literature. The comparative studies direct attention to acceptable practice in the U. S. power reactor industry. The current review articles, however, constitute the major portion of this issue.

All incoming literature (including reports, books, American and foreign technical journals, and transactions) is examined for subjects within our area of interest. This material is collected, grouped, and reviewed by experts. Interpretations, in any article, represent the opinions of the editors, who are employed by the Oak Ridge National Laboratory. Readers are urged to consult references to original work for more complete information.

In addition to the invited contributors, many members of the Oak Ridge National Laboratory staff wrote review material, reviewed manuscripts, or otherwise contributed to this publication. Their contributions are gratefully acknowledged.

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Issued quarterly by the U. S. Atomic Energy Commission. Use of funds for printing this publication approved by the Director of the Bureau of the Budget on June 26, 1959.

Hazards of Nuclear Power Units in Space Applications

Utilization of nuclear energy sources for propulsion and for auxiliary power is essential if many of the proposed space missions are to be accomplished. As power requirements progress into the multikilowatt range for missions of one month or longer, the specific weight of a nuclear power plant makes it clearly superior to other power sources.[1-3] As a result, it may be necessary to live with nuclear energy sources in space.

Consideration of the possible hazards may be divided a number of ways. The hazards to the general public are discussed first here, and then the hazards to the occupants of the space vehicle are assessed. Hazards to the general population will occur only when the vehicle is in the vicinity of the earth's surface, i.e., during the launch phase of the mission or during the return phase, whether planned or unplanned. Attainment of high reliability in the launching and guidance systems and prudent choice of launching site and initial trajectory can, of course, minimize the hazards. The nuclear-energy-source applications currently under consideration for space vehicles include both small reactor power plants and radioisotope energy-conversion plants.

Hazards to Population During Launch

The period of space-vehicle launch may be defined as the time from erection of the vehicle on the launching pad until a satisfactory orbit or trajectory into space has been established. This is probably the time of most serious hazard for a vehicle with a radioisotope energy source since the full load of radioisotopes will be aboard, and the radioactive inventory will be at its maximum.

On the other hand, a space vehicle that utilizes a reactor power plant can be relatively safe during the launch period, since the preflight testing and operating program can leave such a power plant with a relatively low inventory of radioactive fission products. If radiation hazards were the overriding consideration, the reactor could be started after the course of the vehicle had been established and there was a minimum danger of abort or quick return. This solution to the radiation-hazard problem would, however, necessitate the launching of much additional equipment for reactor startup. A compromise solution would be to start up the reactor power plant and establish low-power operation on the ground during the countdown period before launch. The system could then be brought to full power during the final moments of the countdown or very soon after the flight course was established. This alternate startup procedure would probably be preferable because full heat dissipation from the radiator would probably not be practical until the aerodynamic fairing protecting the radiators had been jettisoned. Fission products would build up in the reactor during testing and startup, and the allowable fission-product inventory before launching would be determined as a function of the launching site, reactor type, etc. Should the reactor exceed the allowable radioactivity level because of countdown holds or power excursions, the shot could be called off.

An evaluation of the hazards with respect to the surrounding population, the operating personnel, and the facility will be necessary for prelaunching operation of the nuclear power plant, as well as for abort and reentry conditions.

A launch abort in which the nuclear system remained intact would present problems that would vary according to the location of the landing. Recovery or cleanup operations would be required, and, depending on the past power history and whether the package fell into water or onto ground, the cleanup operations would be relatively easy or difficult. The power plant will probably be shielded over only a small solid angle, and therefore intense radiation fields will have to be considered even if the package remains intact. Recovery will be accomplished either with equipment remotely controlled through the use of closed-circuit television or with heavily shielded, directly controlled equipment. Underwater recovery might be somewhat simpler than ground recovery if the power-plant package remained intact and subcritical, since the water would provide a certain amount of shielding. Should the structure not remain intact, the further problem of water contamination would exist.

If the power-plant package did not remain intact during a launch abort, the hazards would depend on the time and cause of the abort. An abort on the launching pad might involve burning of the nuclear energy source in the rocket blast, whereas an abort at higher altitudes might lead to high-explosive dispersal of the power-plant components.

Hazards to Population During Return

The vehicle may return to the earth's surface either prematurely as a result of some malfunction or at the conclusion of a scheduled mission. An unscheduled quick return to the surface might occur in the launching phase because of an undesired trajectory as a result of a guidance-system malfunction, improper engine performance, or some other misfortune. Of the possible objectionable trajectories, probably the most serious would be a short-range path in the earth's atmosphere that did not involve reentry conditions. From nuclear hazard considerations, it would be advantageous to have a low fission-product inventory in this case, since the power plant would probably not land intact, and the debris might cover an area too small for a safe degree of dilution of activity.

For the reentry trajectories, the vehicle could be designed so that a shaped charge of high explosive could effect the initial breakup and reentry heating would vaporize and disperse the fragments. For best results, such a procedure would have to be carried out at altitudes far above the levels used for air transportation. A small particle dispersion is desired to minimize fallout.[4]

In general, at a given power level, the radioisotope energy source will contain a greater amount of radioactive material than a reactor during the initial part of a mission. As the mission progresses, the radioisotope inventory will decrease, whereas the reactor fission-product inventory will build up to some level, probably surpassing the radioisotope inventory, and maintain a substantial level throughout its useful life. An attempt to assess the relative hazard of the radioactive inventory is difficult since both the reactor and the radioisotope source may take on a variety of forms, and a number of parameters must be considered, e.g., containment of the reactor is limited by requirements for critical configuration and control, whereas the isotope can be of a number of shapes and no control is necessary on the isotope package. Radioisotope power units are best suited to missions requiring less than 10 kw of electrical power, whereas the reactor plant, because of critical size restrictions, cannot compete favorably until the kilowatt power range is reached. At the higher power levels, the reactor is clearly superior.

The disposal of the nuclear energy source after a mission will present problems somewhat similar to those associated with the unexpected trajectory. The situation will be radically more hazardous, however, because of the buildup of fission products. A suitable cooling period could be scheduled to allow the short-lived fission products to decay before destruction of the nuclear energy section in the upper atmosphere. A more attractive solution would be simply to boost the nuclear energy system into deep space with a reserve rocket.

Hazards to Space-vehicle Occupants

Here consideration is given only to the hazards to the space-vehicle occupants as a result of the presence of a nuclear power plant since malfunctions of other components could occur without affecting the reactor. Of the hazards not attributable to malfunctions of equipment, close approach to any scattering material will probably present the major hazard for the vehicle occupants. Since highly asymmetric

shielding is being considered for space-vehicle nuclear power plants, the hazard from reflected radiation exists for the vehicle occupants both during countdown procedures and during landing on another body with a significantly reflecting atmosphere. Atmospheric scattering could drastically increase dose levels. Any close approach to another vehicle in space would pose additional problems since there would be the possibility of rescattering from the neighbor's ship, as well as a direct dose from his power plant. A rendezvous of vehicles containing nuclear energy sources would require careful alignment of the vehicles and would complicate already difficult approach problems.

In certain of the space power designs, a fission-product leak from a fuel rod could also be dangerous to the space-vehicle occupants. Simple crew-compartment shielding may not be the answer to these problems since such shielding could increase the crew dose from cosmic radiation. An evaluation of the magnitude of the cosmic-ray hazard is to be undertaken at the Oak Ridge National Laboratory (ORNL) in the near future.[5] (P. G. Lafyatis)

Evaluation of Nuclear Safety

The question of standards for evaluating the safety aspects of the use of radioactive materials has been considered in previous issues of this Review,[6–8] and in each instance it has been concluded that there is a need for codification of the criteria for evaluation. In this regard it is pertinent to comment on the opinions of C. Rogers McCullough, member and former chairman of the AEC Advisory Committee on Reactor Safeguards (ACRS). McCullough points out in a recent paper[9] entitled "What Price Safety?" that reactor operations always involve some finite risk. The question then arises of whether the risks are worth the gain. Since there is insufficient evidence to evaluate the risks realistically on the basis of experience, the argument is advanced that *if* the number of reactor accidents is not more than one per year and *if* each accident on the average is the same as that assumed in the "Brookhaven Report"[10] (a 500-Mw reactor with a containment vessel that lets 0.1 per cent of the total fission products escape), the whole population of the United States would on the average receive less than 1 per cent more radiation than they would normally receive. McCullough concludes that under these conditions the gains from peaceful use of atomic energy exceed the risk.

Several points were also made by McCullough regarding protection of the public from radiation. First, radioisotopes are more widely used every day, and the possibility of a menace to the public as a result of their use is becoming more real. The supervision of radioisotope handling procedures should become increasingly a matter of localized control as opposed to federal control; specifically, it should be a local responsibility with the cooperation of the U. S. Public Health Service. Second, there are relatively few people who have the required knowledge and technical background to assess the hazard from radiation damage resulting from misoperation of reactors, and, in general, these people are connected with the AEC in some way. Since the number of high-power reactors will remain small for some time, it seems logical that the individual states should not attempt to handle reactor hazards problems with their own staffs.

The AEC has a legal responsibility for the health and safety of the public. It uses the following procedure for determining the adequacy of safeguards in power and test reactors: a review by the ACRS is required; a report of this review is made public; and a public hearing is held. The final decision as to the safety of the facility is made by the Commission. In discussing methods of evaluation, McCullough said that the ACRS review consists of an examination to see what safety margins have been provided in the basic design, the control system, and the operating procedure. The reviewers do not necessarily check the design and specifications of each individual part, but they must determine that codes or standards are met. Where there are no standards, the reviewers must be assured that the designer has provided an adequate margin of safety.

McCullough stated that at present some unwritten criteria are used in evaluating reactors. The need for codification of criteria for reactor sites and specifications grows continually more acute, but at the present state of experience the barest principles that could be included as particulars vary over an impossibly wide range. Criteria written at this time, he feels, could conceivably freeze design safety practices in a detrimental fashion and impede

the freedom of reactor designers to experiment with new concepts and refinements.

The cooperation between the ACRS and the various divisions of the AEC has been invaluable in judgments based on the preliminary criteria now available. It does not lie within the scope of the AEC responsibilities to initiate reactor design, but the AEC staff does review and evaluate the competence and hazards of designs submitted by applicants for licenses, as well as the designs of reactors authorized for construction at AEC and other government-owned facilities. Responsibility for the competence of the design lies with the designer.

McCullough thinks that it is highly desirable to continue the ACRS for the near future. He sees no immediate need for a separate agency to evaluate the hazards of large reactors, and, further, he deems it necessary at this stage to keep the whole problem within the jurisdiction of the AEC—in public health as well as in reactor aspects. However, when large reactors become more numerous, there will be a need for an agency to control their operation and to safeguard the public. It is hoped there will be general agreement on criteria before the creation of such an agency. (C. E. Winters)

Legal Problems in Atomic Energy

In this continuation of reviews[11] of the development of state, national, and international law regarding atomic energy damage and liability, one paper of international[12] and one of national[13] scope are reviewed, and a recent development[14] in state and federal cooperation in law is briefed.

International Law

In probing the question of legal protection against nuclear damage, the International Atomic Energy Agency (IAEA) reports in its April 1959 Bulletin the purpose and progress of an international panel of experts on the question of civil liability and government responsibility for nuclear hazards.[12] The report states that one of the basic objectives of the IAEA has been to ensure that its activities in promoting the peaceful uses of atomic energy do not defeat their own purpose by posing a danger to public health or by harming the interests of the public in any other way. The public naturally expects that in the event of nuclear damage there will be liability on the part of the enterprise which caused the damage. The situation could be complicated, however, by a variety of factors. For example, several parties might be involved in the establishment and operation of the enterprise. The damage might exceed the liability or the financial resources of the enterprise. The extent of the damage might extend beyond national boundaries, and it is possible that the parties involved in an enterprise would belong to different countries.

The report states that the absence of adequate domestic law covering third-party liability and its financial protection has already led to difficulties in bilateral agreements on the supply of fissionable materials and reactor equipment. Any multilateral approach is more seriously affected by such legal uncertainty, and this is bound to retard international participation in the type of scientific and technical assistance which IAEA has been seeking to initiate and organize. The difficulties would increase further if national legislation of the various countries were to incorporate different principles and procedures.

In the interest of international legal harmony in this field, the IAEA panel has set forth certain basic postulates. The first postulate is that the use of nuclear energy should be regulated by adequate licensing and control mechanisms aimed at preventing accidents. To the extent that nuclear damage cannot be certainly prevented, there must be liability on the part of the enterprise which caused the damage, and, where the damage might exceed its legally enforceable liability or its financial resources, there should be some assurance of compensation by the State. Litigation with respect to liability should be concentrated in the most convenient tribunal and be governed by a single clearly defined law. The liability of an enterprise should not exceed its reasonable financial capabilities, which means that a ceiling should be imposed upon the amount of third-party liability to which an enterprise could be held.

The report further states:

> Although the basic postulates appear simple, the specific questions have many complex aspects. For example, the responsibility of States raises the question both of international responsibility and of their general responsibility with respect to damage not privately compensated. In the field of civil liability, the existing laws and procedures often seem insufficient for protecting the public and at the same time reducing the industry's burden of liability to

reasonable proportions. Should the task of revising these basic rules be left to individual States? Again, the question arises whether any rules limiting civil liability and governing State responsibility do not presuppose the existence and enforcement of adequate standards of safety in the construction and operation of nuclear installations. The question of liability cannot be isolated from efforts to prevent accidents through regulatory provisions. Another question to be decided is whether emergency measures after a nuclear accident should be removed from the area of civil liability and left to the State.

These are but a few of the questions to be decided by the panel in the initial stages of its work. Even from a preliminary investigation, other issues would emerge. For example, it is difficult to conceive of any solution in connexion with civil liability and responsibility of States which would not also include transportation of radioactive materials capable of causing damage. Yet transportation poses a number of legal problems, such as selection of the person and State to be held responsible for nuclear damage, etc. Again, the disposal of radioactive waste may call for special treatment in connexion with civil liability and responsibility of States.

National Law

In regard to limitations on governmental liability under the Federal Tort Claims Act for injuries resulting from nuclear explosions, it was reported at the hearing on health and safety problems and weather effects associated with atomic explosions before the Joint Committee on Atomic Energy, 84th Congress, 1st Session 29 (1955), that "five hundred and seventy-one claims have been received by the AEC from the beginning of the Nevada tests in 1951 up to the beginning of the current test series — February 1955. Three hundred and ninety-three of these claims were paid by the AEC. The total amount paid was $52,159.34."

Waaler, in a paper reprinted in the *Atomic Energy Law Journal*,[13] recognizes that recent cases involving damage alleged to have resulted from the planned detonation of a nuclear device by the AEC have focused attention upon the liability of the United States for injuries resulting from such explosions. He points out that under the Anglo-American legal system, the sovereign has traditionally enjoyed immunity from tort liability to private persons injured by the acts of its agents. A departure from the long-honored rule of sovereign immunity was made in the United States by the enactment in 1946 of the Federal Tort Claims Act which waived immunity of the federal government from liability in tort. However, government liability is still subject to several limitations. It is Waaler's contention that actions for damage from a nuclear explosion must be brought under the Federal Tort Claims Act, and he lists and discusses three limitations of governmental liability:

1. One requirement in the statute is that the action be based upon a negligent or wrongful act or omission on the part of a government employee.

> It is arguable that this requirement is unwarranted and that the government ought to be strictly liable for all harm proved to have resulted from such explosions.

The author's justification for this position is found in the leading case of Rylands vs. Fletcher, which expressed the doctrine that a defendant is liable without fault for harm resulting from an ultrahazardous activity because such activity exposes the community to an undue risk of harm even if every precaution is taken.

> If the doctrine of liability without fault is to be applied to private persons, there appears to be no plausible reason why the government should be excused from similar liability, unless it is the fear of the staggering cost which might arise in the event of a nuclear "accident." Even in the event of such a disaster, however, the cost would rest more fairly upon the government, where it could be redistributed to the benefactors of the activity, the country as a whole. Notwithstanding the force of this argument, it seems unlikely that the negligence requirement will be discarded in the near future.

2. The second limitation grew out of an interpretation of the Act which limited the government liability to situations where a private person would be liable. It is apparent that some governmental activity has no counterpart in the affairs of private persons. This limitation has recently been rejected by the Supreme Court. In Rayonier Incorporated vs. United States [352 U. S. 315, 77 Sup. Ct. 374 (1957)], the Court said:

> We expressly decided ... that the United States' liability is not restricted to the liability of a municipal corporation or other public body and that an injured party cannot be deprived of his rights under the Act by resort to an alleged distinction, imported from the law of municipal corporations, between the government's negligence when it acts in a "proprietary" capacity and its negligence when it acts in a "uniquely governmental" capacity.

It would seem that the liability of the government is to be determined by the same rules as the liability of a private person, regardless of the objective of the function.

3. The third limitation is stated under section 2680(a) of the Act where it is provided that, even though negligence is proved, no liability can be based on "the failure to exercise or perform a discretionary function or duty on the part of a federal agency or an employee of the government, whether or not the discretion involved be abused." The author cites cases where the government has been held immune from liability if some element of discretion was exercised by a government employee. Subsequent cases are cited, however, and they show a tendency on the part of the courts to take a contrary view. In the author's opinion,

> The rationale of the discretionary function exception is that in order to effectively exercise the judgment vested in them, governmental officials must be allowed freedom of choice without fear that possible consequences of their acts will subject the government to liability. Thus the courts have held that acts which involve a certain amount of judgment must carry with them immunity from liability. Counterbalanced against the principle of free exercise of discretion is the apparent justice of compensating innocent persons injured as a result of governmental activity. It is doubtful that in most cases the relatively slight restraint upon the exercise of discretion represented by the imposition of legal liability for consequential harm outweighs the justice of compensation.

(David G. Schultz)

Regulatory Jurisdiction of States

Prior to the 1954 amendments to the Atomic Energy Act of 1946, ownership of all nuclear materials and reactors was vested in the federal government. The 1954 amendments granted, under prescribed conditions, the right of private organizations and individuals to possess nuclear materials and to own and operate nuclear reactors. Thus private companies and individuals were permitted to work with nuclear energy in private capacities under licenses issued by the AEC instead of operating as agents of the AEC.

In September 1956, the Council of State Governments, at a meeting in Washington, suggested a uniform statute to provide general coordination between the states and the AEC in the field of atomic energy. The various states formed study groups and enacted state legislation in the atomic energy field, and, in 1957, the AEC proposed a specific bill to Congress for transferring certain federal regulatory powers to the states. The Commission later modified the proposed bill and resubmitted it to the Joint Committee on Atomic Energy in May 1959. The modified bill was passed and, on Sept. 23, 1959, became Public Law 86-373. This law further amended the Atomic Energy Act by adding Section 274, entitled "Cooperation with States," and provided for a far-reaching change in the legal status of state activities involving nuclear materials. It made some of the nuclear activities subject, for the first time, to state control and regulation.

The 1959 bill differed from the 1957 bill in that it provided procedures and criteria whereby the Commission could turn over to the individual states, as they became ready, certain definite areas (described below) of regulatory jurisdiction. Congress specifically provided for retention of Commission authority and responsibility for regulation of "the construction and operation of any production or utilization facility" and limited the release of federal jurisdiction in definite areas. Areas in which interstate, national, or international considerations might be paramount were excluded. Some areas were excluded because the technical safety considerations were of such complexity that it was not likely that any state would be prepared to deal with them during the foreseeable future. The Commission was also empowered to terminate or suspend such agreements as required to protect public health and safety.

The major provisions of Public Law 86-373 are presented in the following abstracts and excerpts:

Paragraph *a* states that the purposes of Section 274 are (1) to clarify respective responsibilities of the states and the Commission with respect to the regulation of by-product, source, and special nuclear materials; (2) to establish programs for cooperation between the states and the Commission with respect to control of radiation hazards; (3) to promote an orderly regulatory pattern between the Commission and the state governments with respect to nuclear development, use, and regulation of by-products, source, and special nuclear materials; (4) to establish procedures and criteria for discontinuance of certain of the Commission's regula-

tory responsibilities and the assumption thereof by the states; (5) to provide for coordination of the development of radiation standards for the guidance of federal agencies in cooperation with the states. It was recognized by Congress that, as the states improve their capabilities to regulate such materials effectively, additional legislation may be desirable.

Paragraph *b* authorizes the Commission to enter into agreements with the governor of any state to provide for discontinuance of the regulatory authority of the Commission with respect to (1) by-product materials, (2) source materials, and (3) special nuclear materials in quantities not sufficient to form a critical mass.

Paragraph *c* provides the limitations in the proposed agreements by stating:

> No agreement entered into pursuant to subsection *b* shall provide for discontinuance of any authority, and the Commission shall retain authority and responsibility with respect to regulation of (1) the construction and operation of any production or utilization facility; (2) the export from or import into the United States of by-product, source, or special nuclear material, or of any production or utilization facility; (3) the disposal into the ocean or sea of by-product, source, or special nuclear waste materials as defined in regulations or orders of the Commission; and (4) the disposal of such other by-product, source, or special nuclear material as the Commission determines by regulation or order should, because of the hazards or potential hazards thereof, not be so disposed of without a license from the Commission.
>
> Notwithstanding any agreement between the Commission and any State pursuant to subsection *b*, the Commission is authorized by rule, regulation, or order to require that the manufacturer, processor, or producer of any equipment, device, commodity, or other product containing source, by-product, or special nuclear material shall not transfer possession or control of such product except pursuant to a license issued by the Commission.

Paragraph *d* states the conditions that must be met to enable the Commission to enter into an agreement with a state.

Paragraph *e* provides that a proposed agreement must be published in the *Federal Register* once each week for four consecutive weeks to provide an opportunity for comment by interested persons.

Paragraph *f* authorizes the Commission by regulation or order to grant certain exemptions from the licensing requirements set forth in the Atomic Energy Act of 1954, as amended.

Paragraph *g* authorizes and directs the Commission to cooperate with the states in the formulation of standards for protection against hazards of radiation.

Paragraph *h* establishes a Federal Radiation Council to advise the President with respect to radiation matters, directly or indirectly affecting health, including guidance for all federal agencies in the formulation of radiation standards and in the establishment and execution of programs of cooperation with states.

Paragraph *i* authorizes the Commission to perform inspections or other functions on a cooperative basis with the states and authorizes the Commission to provide training with or without charge to employees of the states and political subdivisions thereof.

Paragraph *j* states that the Commission may, upon its own initiative, after reasonable notice and opportunity for a hearing, terminate or suspend an agreement with a state if the Commission finds that such termination or suspension is required to protect public health and safety.

There appears to be a gray area of divided jurisdiction between the state and federal governments with respect to air or stream pollution and waste disposal by an AEC contractor operating in federally owned facilities. This question of jurisdiction was discussed in the hearings. It was decided not to include a specific delineation between the jurisdiction of the federal government and the states; rather, it was decided that it would be better for the Commission to review the states' proposed regulations prior to entering into an agreement with them to ensure that the federal regulations governing a federally owned facility were no less stringent than the proposed state regulations. Essentially, this would place the contractor operating the federally owned facilities in the same position with respect to state regulations on radiation hazards, etc., as with respect to boiler inspection regulations, etc.

The new law would seem to place the burden on AEC contractors in their relationship with subcontractors, AEC contractors of nonfederally owned facilities, and radioisotopes customers to determine, as they now do, not only whether a license is needed but whether the parties have complied with the various state regulations where the states have assumed regulatory powers. (L. A. Mann, based on notes prepared by the Legal and Information Control Depart-

ment of the Union Carbide Nuclear Company, Union Carbide Corporation, acting as a management contractor for the U. S. Atomic Energy Commission)

Safety in Nuclear Ship Propulsion

International acceptance of the use of nuclear power in commercial shipping is a vital prerequisite to its widespread economical use in world trade. Such acceptance is, of course, contingent upon demonstration of the reliable and safe operation of nuclear-powered ships. In this application of nuclear power, unique problems result from the need for small specific volume and weight; from the relative proximity of ship personnel to the radiation sources; from the stresses arising from the motion of the ship; from the requirements for nuclear safety in the event of collision, sinking, or grounding; from restrictions in the locations and circumstances in which disposal of wastes may be allowed; and from the incentive to recover and salvage expensive fuel from a damaged or sunken vessel.

As an initial step in the resolution of these problems, Atomic Energy Panel M-13 was organized in 1955 under the sponsorship of the Ships Machinery Committee of the Society of Naval Architects and Marine Engineers. At the request of the U. S. Coast Guard, the panel analyzed the problems and established criteria for safety in the design, operation, and maintenance of a nuclear-powered merchant ship.[15] The application of these criteria in analyses of the safety features of the N.S. *Savannah* is discussed here, together with the results of an analysis of the effects of ship accidents[16] on the power plant.

In addition to these discussions that are specific to the N.S. *Savannah*, a review is presented of preliminary studies that were made by the Armour Research Foundation[17–23] on scavenging radioactivity from water and air. A number of general statements of proposed methods of solution of the problems of nuclear propulsion of merchant vessels have appeared in the American and British literature and are listed here for reference but are not reviewed.[24–28]

Safety Features of the N.S. Savannah

The N.S. *Savannah* is, of course, the only nuclear-powered ocean-going vessel that has been thoroughly studied with respect to major safety aspects.[29] The studies have included those of Babcock & Wilcox,[30–33] the power-plant designer; George G. Sharp, Inc.,[16] the ship designer; and ORNL.[34]

Credible failures of conventional components of the ship or its machinery and equipment that would adversely affect nuclear performance or safety were considered in the development of the design. Conservative factors of safety were used with respect to normal operating stresses and hazards and possible accidents, such as collision, grounding, flooding, fire, storm, explosion, sinking, foundering, and capsizing.

The pressurized-water reactor and its auxiliaries and protective surroundings are designed for dependability and safety both in normal operation and in all credible accident situations. The system is provided with all the containment safeguards applicable to land-based reactors, and, in addition, the reactor and boilers and, to the extent feasible, the reactor and boiler auxiliaries are enclosed in an unusually compact and heavy steel containment vessel located in the geometrically safest part of the ship. Detailed attention has been given to the safety of all features of the containment vessel, including the required penetrations for fluids and instrument leads through the vessel wall. The penetrations for circuits of the primary cooling system will carry only cooled and depressurized bypass water flowing out to purifiers and repressurized purified water flowing in from the recompressors. The recompressors are contained within the heavily built and shielded compartment that surrounds the entire nuclear power complex. The shielded compartment is maintained at subatmospheric pressure and ventilated by air flowing at 4000 cfm. The air is mechanically and chemically filtered before being discharged from the stack. The compartment shielding consists of a matrix of concrete, lead, and plastic. The ship's bottom is a double shell with a compartmented annulus between the shells. The sides of the ship are protected from deep collision penetration by laminated steel and redwood buffers. Analyses show the chance of damage to the nuclear plant from collision or grounding penetration to be essentially nil in harbors and inland waterways, and practically, although not absolutely, incredible on the open seas.[16,30]

In order for fission products to reach the atmosphere, they would have to escape from

the fuel, through the fuel can into the primary cooling water, through the coolant-container wall into the reactor containment vessel, through the containment-vessel wall into the compartment, through the mechanical and chemical filters, and out the stack. Similarly, activated corrosion products and products of nuclear reactions with the coolant and its impurities would have to follow the same path, with the omission of the first two steps. Inspection and test requirements restrict the maximum leakage rate from the containment vessel into the compartment, and thus to the stack, to less than the minimum detectable rate, estimated to be about 0.2 per cent per day.

All radioactive liquid and solid wastes are to be stored in the ship and disposed of only when the ship is in a port specifically equipped to handle the wastes. Radioactive gaseous wastes will be continuously monitored and will not be discharged when the combination of activity, weather, and environment could credibly cause a dosage that would be above that recommended by the International Commission on Radiological Protection (ICRP).[35]

Conceivable accidents and malfunctions were analyzed for consequences and credibility.[32] Those which were found to be of significant consequence and to be credible were further analyzed, and protective features were incorporated in the design to reduce the probability. The maximum credible accident was considered to be the series of events which might follow failure of the primary coolant piping, since this could be followed by meltdown of the fuel and dispersion of fission products inside the containment vessel. The possible consequences of this maximum credible accident were analyzed in terms of the nuclear hazards involved.[32–34] The analyses indicated that the most significant requirement in terms of safety is that the containment vessel *not* be opened, such as for inspection or maintenance, while the nuclear afterheat potential is great enough to cause rapid failure of any fuel element in case of loss of primary coolant while the vessel is open.

The hazards analyses of the N.S. *Savannah* are largely applicable to other types of vessels and to other types of nuclear power plants for vessel propulsion. It is claimed that, in so far as has been practically determinable, the N.S. *Savannah* has been designed for maximum feasible safety and dependability. Testing and operating experience with the ship will doubtless disclose many possible technical improvements, as well as many reasonable economies which can be applied to subsequent vessels. Not only are studies of improvements in progress, but many other studies and test programs pertinent to nuclear ship propulsion are under way.

Environmental Analysis of N.S. Savannah Operations at Camden

The over-all conclusion of an extensive analysis[34] of the hazards to the environment from operation of the N.S. *Savannah* at Camden, N. J., was that the maximum credible accident could not result in doses as high as the maximum acceptable doses recommended by the ICRP[35] from direct radiation, atmospheric contamination, or water contamination. The ACRS has not, however, completed its review of the analysis, and it is possible, for example, that even the small reliance on administrative control to assure that the containment vessel not be opened at other than specified times may not be accepted, although it is argued that similar situations have been found to be acceptable for other reactors.

In the analysis of the environmental hazards, the amount of each fission-product isotope available for release to the atmosphere at the time of the maximum credible accident was calculated from the following formula:

$$Q = Q_0 e^{-\lambda t} F_1 F_2 F_3 F_4 F_5 F_6 F_7 F_8$$

where Q = discharge of the isotope from the stack into the air, either as a rate of continuous discharge or as a total amount discharged

Q_0 = total amount of the isotope existing in the fuel at shutdown ($t = 0$)

t = time elapsed since reactor shutdown

λ = fractional rate of decay of the isotope

F_1 = fraction of Q_0 that has escaped from the fuel at time t

F_2 = fraction of $Q_0 F_1$ that has escaped from the fuel cans into the primary coolant system

F_3 = fraction of $Q_0 F_1 F_2$ that has escaped sedimentation and sorption in the primary coolant system and has

moved from the primary coolant system into the containment-vessel free volume

F_4 = fraction of $Q_0F_1F_2F_3$ that has escaped sedimentation or sorption in the containment vessel; $Q_0F_1F_2F_3F_4$ = total amount of isotope suspended in the gas inside the containment vessel

F_5 = fractional leakage rate of gas in the containment vessel from the vessel into the surrounding compartment, including any filtration effect in leaking through the vessel wall; estimated to be 2×10^{-8} sec^{-1}, based on the leakage specification of 0.2 per cent per day^{-1}

F_6 = fraction of $Q_0F_1F_2F_3F_4F_5$ not sedimented or adsorbed out of the gas in the compartment before reaching the stack filters

F_7 = $(Q_0F_1F_2F_3F_4F_5F_6) \div$ (flow rate of ventilation air); F_7 is the dilution effect of the ventilating air; estimated to be $(2 \times 10^{-8} \times 35{,}000$ cu ft) $\div$ [4000 (cu ft)$(min^{-1}) \div 60$ (sec) $(min^{-1})] = 1.05 \times 10^{-5}$, dimensionless

F_8 = fraction of $Q_0F_1F_2F_3F_4F_5F_6F_7$ passing through the stack filters = 1 − fraction removed by filters

The dilution of the effluent radioactive isotope by mixing with the air was calculated[34] by the use of Sutton's atmospheric dispersion equation,[36] which relates dispersion to meteorological conditions. The dose to the population for each important isotope in the event of a maximum credible accident was approximated by Burnett's calculational method,[37] and the total integrated dose to the total population was estimated to be 4060 man-rem,[34] which compares favorably with proposed acceptable risks of 2 million man-rem.[34,38] The maximum total integrated dose to an individual at the worst off-site location was estimated to be 202 mrem. In comparison, the National Committee on Radiation Protection (NCRP) recommends not more than 5000 mrem per year.

In analyses of the contamination of the hydrosphere that would result from submergence of the N.S. *Savannah* in the Delaware River as a part of an unspecified incident leading to a maximum credible accident, predictions of maximum river-water activity concentrations were compared with the dosage recommendations of the NCRP and the ICRP.[34,39] Data from model tests of the Delaware River estuary carried out at the U. S. Army Waterways Experiment Station, Vicksburg, Miss., in 1952 and 1958 were used for the analysis.[40,41] In the model tests, controlled injections of dye, together with controlled programs of water sampling at points and times of interest, were carried out. Although the model test results are directly applicable only for the regimes tested[34] (river flow, tide conditions, programs of injection, location of injection points, and programs and locations of sampling), rough interpolations of sufficient accuracy for the N.S. *Savannah* analyses could be made.

It was assumed that the leakage rate from the submerged containment vessel would not be less than from the unsubmerged vessel, although it was noted that the extra cooling and external pressure on the submerged vessel would make this a conservative assumption. Several different leakage regimes were assumed for each model pattern, and the model data were normalized to the assumed leakage rate, i.e., a ratio factor of 1.67×10^{-14} for the 1958 model tests. The final calculated results are presented as tables and curves of ratio of river-water concentrations to maximum permissible concentrations[39] (MPC) for drinking water, expressed as numbers of MPC units of the significant radioisotopes. For example, 4×10^{-5} μc of I^{131} per cubic centimeter would be expressed as 2 MPC for thyroid tolerance.[33]

The maximum ratio found was less than two times the thyroid MPC for continuous exposure. This maximum occurred near the point at which the ship sank and was at the peak time of a time-dependent curve; it was not near a city water supply inlet. The maximum concentration found opposite the Torresdale (Philadelphia) municipal water intake was less than 0.08 MPC for the most critical organ (in this case, the bone). The results were similar for submersion exposures, which would be less significant because fewer people would be involved.

(L. A. Mann)

Analyses of Ship Accidents

George G. Sharp, Inc., analyzed the nonnuclear hazards to which the N.S. *Savannah* will be exposed during operation, including collision, grounding, heavy weather, fire, explosion, flood-

ing, and sinking.[16] These hazards were considered principally from the standpoint of release of radioactive material rather than of safety of life at sea in the conventional sense. The special hazards of war service, such as operation with naval vessels and ballistic and mine damage, were not considered.

All aspects of conventional ship safety were, however, investigated. The design standards, emergency propulsive power, steering gear, hull structure, stability, visibility, navigation and communication equipment, and maneuvering capabilities of the N.S. *Savannah* were examined and compared with those of other modern merchant ships. It was concluded that, for general safety of operation at sea, the N.S. *Savannah* is designed to the highest current standards and is as safe as, or safer than, any comparable vessel. Because collisions are usually the result of errors in judgment on the part of responsible personnel, they cannot be eliminated through design. Therefore provisions must be included in a nuclear-powered ship to protect the reactor and its containment vessel from the effects of collision. In the investigation, a study of available records on approximately 60 major collisions was made in order to determine the mechanisms of collision, to define the magnitude of damage to be expected, and to develop a basis for the design of collision protection in nuclear ships. It was found that ship collisions are highly inelastic and that the accelerations cannot, under any circumstances, be expected to exceed about 1 g. It was also found that a fairly good correlation exists between the kinetic energies of the colliding vessels and the structural damage incurred.

From the study and consideration of design features of the N.S. *Savannah*, it was concluded that there is no danger of shock damage to reactor components in a collision and that rupture of the containment vessel in a harbor is practically incredible when considered in the light of low harbor speeds and the infrequency of collisions. Therefore the possibility of rupture of the containment vessel in collisions was limited to areas other than harbors, and the probability of such an event during the 20-year life of the vessel was found to be extremely low, of the order of 10^{-5}. The general conclusion was that the possibility of collisions is a fully acceptable risk.

Grounding accidents were considered along the same lines as collisions but in a more qualitative manner. It was concluded that the probability of the N.S. *Savannah* running aground is small and that the structure is such that damage to the containment vessel would be unlikely even if the most dangerous possible sequence of events in the accident led to the complete breakup of the ship. If the containment vessel broke loose, its weight (approximately 700 tons of structure and 600 tons of bonded lead) would cause it to sink.

Scavenging Radioactivity from Water

In considering the possibility of the accidental release of activity in an uncontrolled manner to the confined waters of a harbor in a heavily populated area, it is realized that the consequences could be troublesome, expensive, and perhaps hazardous. Therefore it would be desirable to have remedial measures available which could be employed following such accidental release of activity and which would influence the dispersion of the activity in such a manner that the consequences of the release would be diminished. Studies of the problem are being carried out by the Armour Research Foundation under an AEC contract.[4,10] The research has consisted of the investigation of scavenging, flocculation, and settling of radioactive materials in the liquid phase and of scavenging of radioactive particulate matter in the gaseous phase.[17–23] The emphasis has been on the liquid-phase studies, but some attention has been given to the control of activity released in the gaseous phase, principally to means for tracking radioactive gases which are deliberately released to the atmosphere as part of a normal operating procedure.

The method which is being studied for scavenging radioactivity in the liquid phase consists of sowing the infected water area with chemicals which disperse in a region extending to perhaps 10 m below the surface to form a precipitated scavenging medium, or floc. The radioactive materials present in solution or as a hydrosol become attached to the scavenging-agent flocs and are carried to the bottom in a sedimentation process. Thus activity is transferred from the liquid phase to a solid phase where it is presumably less hazardous. The presently recommended chemical agents are potassium permanganate and anhydrous ferrous sulfate pellets which react to form a manganese dioxide–ferric oxide hydrate scavenging medium. An additional

neutralizing agent may be required to maintain a favorable *p*H. The research work reported has included laboratory studies of proposed scavenging agents and of means for manufacture and distribution of pellets. The work has been reported in two rather detailed reports[17,18] and in brief progress reports.[19–23]

In laboratory-scale experiments[17] it was found that more than 90 per cent of most of the fission-product radionuclides used were removed from solution within several hours, using $KMnO_4$ and $FeSO_4$ scavenging systems. Notable exceptions were strontium and cesium which were scavenged only slightly or not at all. Preliminary tests of other methods for removal of strontium and cesium were performed. Although no highly effective methods were demonstrated, it was felt that some agents were sufficiently promising to merit further investigation. Other preliminary experiments indicated that the radionuclides adsorbed by the manganese dioxide–ferric oxide hydrate were strongly attached and that desorption did not readily take place.

The sedimentation characteristics of similar concentrations of potassium permanganate with either ferrous chloride or ferrous sulfate were compared directly.[17] Larger flocs and faster settling particles formed with the ferrous sulfate system. Particle size and hence settling velocity increased with potassium permanganate concentration up to but not beyond about 50 ppm. A preliminary field test program indicated that saturating a given volume of sea water with scavenging agents would give better results than forming flocs only on the surface of the water and letting them settle.

The behavior of crystals of scavenging chemicals as they fall through sea water was investigated, especially their dissolving and settling rates.[17] The results were used to estimate the mass of a crystal which would just dissolve in falling through 10 m of sea water and the resulting concentration at each depth. The calculated concentration varied with depth only by a factor of 2 over all but the last part of the crystal's fall. The experiments indicated, however, that the applicability of these calculations to sinking crystals is somewhat questionable if precipitate formation occurs, as with potassium permanganate.

A dispersing system based on the use of pelleted chemicals rather than crystals was investigated.[17] The goal was to make pellets which would dissolve uniformly while falling through 10 m of sea water to give any desired concentration of the chemical. Fabrication procedures were developed for making pellets of individual chemicals, and disintegration tests were made by allowing pellets to sink through 10 m of simulated sea water in a 6-in.-diameter Lucite tower. The disintegration rate was reported to be uniform along most of the tower length, except in the first meter of depth where wetting was taking place.

Some consideration has been given to requirements for storage of scavenging chemicals on board a nuclear ship and to methods of using them when needed.[17] Although apparently no experimental work has been done, the evident importance of applying the scavenging agent near the source of contamination is stressed. The amounts of scavenging agents required have been estimated at 15 metric tons of $KMnO_4$, 45 each of $FeCl_2$ and $2H_2O$, and, if used, 10 of sodium orthosilicate neutralizer. A pneumatic conveying system is suggested as the most feasible system for delivering the materials in pellet or granular form. A suggested distribution pattern is to cover an area on each side of the ship 50 by 100 ft at rates up to 120 tons per hour.

Any attempt at this time to evaluate this research work in terms of nuclear merchant ship safety would be premature. Although the desirability of achieving a means for limiting the spread of activity accidentally released to sea water seems fairly clear, additional research work will be required to establish the effectiveness of proposed scavenging techniques. Certainly, procedures which are ineffective for strontium are of questionable merit, and some modification of the presently proposed potassium permanganate–ferrous sulfate scavenging system may be anticipated. Measurements of scavenging effectiveness have not included some fission products, such as iodine, which may have an important or even controlling influence on hazard aspects of an accidental activity release. Tests which were conducted to determine the scavenging and sedimentation properties of chemical agents were carried out under laboratory conditions in which distribution, mixing, contacting, and settling conditions were much more closely controlled than would be possible in a field application. Procedures and equipment for distribution and delivery of scavenging agents to the infected water area

are only in the conceptual stage of development. It appears that tests of the scavenging system under conditions of anticipated usage are necessary before the usefulness of this potential tool can be established, and plans for such tests are included in the research program.[23]

Scavenging Radioactive Particulate Matter from Gases

Little work has so far been done on the scavenging of radioactive particulate matter from gases, possibly because situations in which this technique could be usefully applied to nuclear merchant ships are not immediately obvious. Early in the research program of the Armour Research Foundation, a series of calculations was made to examine the behavior of a cloud of finely divided radioactive matter released from a nuclear ship.[18] Sutton's equation, with Chamberlain's correction for deposition,[17] was used for four sets of parameters. The results were presented in a series of 76 graphs. It was intended to utilize these calculations in the analysis of results obtained from scavenging experiments in the gaseous phase, but apparently such experiments did not materialize. The graphs might be useful, however, in generalized studies of spatial distributions of particulate matter.

Attention has been given to means for tracking radioactive gases which may be periodically released to the atmosphere from a nuclear ship as part of a normal operating procedure.[17] The objective is, of course, to prevent the gas plume from drifting into undesired areas. It is proposed that the trajectory of the gas plume be predicted by tracking a simultaneously released target visible to the eye or to radar. The investigations reported have not yet included experimental evaluations of techniques.

One tracking method being considered is the release of a visible smoke to follow closely the path of released radioactive gases. The characteristics of readily available smoke generators were investigated, and only the Army Chemical Corps HC smoke pots appeared to have promise for this application. Some proposed radar tracking experiments were analyzed, and it was concluded that constant-pressure balloons were the most favorable targets for this application. Calculations of radar reflectivity indicated that balloon targets, either 2-ft-diameter spheres or 3-ft tetrahedrons, could be detected at satisfactory distances. (H. C. McCurdy)

Changes in Financial Protection Regulations

As the industrial use of atomic energy becomes more widespread, the potential for damage to persons and property from the release of accumulated radioactivity increases. The accidental release of radioactivity has fortunately been limited in the past, and thus there is no extensive experience record on which to base a realistic evaluation of probable population exposures and land contamination in the event of an accident at a particular installation. As in other industrial pursuits, the industrialist is responsible for the protection of the public, and he must balance the financial gain from productive operation against the cost of insurance or possible losses.

The insuring organizations must attempt to set rates commensurate with the risk and a fair profit. The historical and modern background of insurance and its relations to people, industry, and atomic energy was reviewed in the previous issue of this Review.[42]

The government has the double duty of protecting the public against biological and financial damage and, at the same time, of furnishing the public the opportunity of benefiting from all reasonable technological advances. Federal law, in addition to requiring the executive branch to guard the safety of the public in the development and use of atomic energy, also specifies a framework within which the AEC controls the amount of private financial protection required of users of atomic energy and handles the government indemnification of damages in excess of the required private protection. In September 1957, the Commission issued regulations[43] specifying detailed requirements for private financial protection for atomic power reactors. Since that time, it has continuously studied possible improvements to those regulations, has invited comments and suggestions relative to them, and has met with interested parties and organizations to hear their comments and proposals. Based on these studies the Commission published,[44] on Apr. 1, 1960, amended regulations for determining the amount of private financial protection required of a licensee as a prerequisite to (1) possession and storage of

fissionable material prior to its use in a reactor and (2) operation of a testing or power reactor. The amendments were effective May 1, 1960. When the financial protection regulations were originally issued in September 1957, the Commission stated that they were not intended to be final, that changes to more definitive regulations would be made, that other types of provisions would later be included, and that the amounts of financial protection required by the more definitive regulations might vary substantially from the September 1957 requirements. The amendments to the regulations which the Commission has now issued are intended to adjust the amounts of private financial protection required to more closely correlate with the risks involved. Essentially, the amendments are intended to relate the required financial protection to the population distribution in the surrounding area, the thermal power level of the reactor, and the dispersion of radioactivity in the atmosphere.

The amended regulations increase the minimum amount of protection required from the previous value of $250,000 to the new value of $1 million; they fix the amounts of protection required for small [10 Mw(t) or less] reactors without reference to population density; and they increase the protection required for reactors with power levels between 10 Mw(t) and 100 Mw(e) as a specified function of population distribution in the surrounding geographic area.

The amended regulations continue the $60 million maximum private financial protection for reactors licensed to produce 100 Mw(e) or more, as an interpretation of Public Law 85-256, which requires that the total available amount of private liability insurance be furnished for each reactor in that category. The provision for government indemnity to $500 million liability above the private protection provided is also retained unchanged. An annual fee is charged by the Commission for this protection, and the fee amounts to $30 per authorized megawatt of thermal power for the largest reactor at a location, with the minimum fee being $100 per annum. The seller of a reactor of less than 3⅓-Mw(t) capacity is charged a fee of $100 per reactor if the reactor is operated at the seller's location prior to sale. The Commission is authorized to exempt certain categories of reactor operators from financial protection requirements if the specific conditions set forth in the regulations are fulfilled.

For reactors that are to be operated at power levels less than 100 Mw(e), the licensee is required to have and to maintain financial protection

1. In the amount of $1,000,000 for each nuclear reactor he is authorized to operate at a thermal power level not exceeding ten kilowatts;
2. In the amount of $1,500,000 for each reactor he is authorized to operate at a thermal power level in excess of ten kilowatts but not in excess of one megawatt;
3. In the amount of $2,500,000 for each nuclear reactor other than a testing reactor [defined in reference 44] or a reactor licensed under Section 104 B of the Act which he is authorized to operate at a thermal power level exceeding one megawatt but not in excess of ten megawatts; and
4. In the amount of $60,000,000 for each nuclear reactor he is authorized to operate and which is designed for the production of electrical energy and has a rated capacity of 100,000 electrical kilowatts or more.

For reactors whose licensed capacity exceeds 10 Mw(t) but is less than 100 Mw(e), the following formula is used to determine the financial protection required, subject to the minimum and maximum private protection requirements of $3.5 million and $60 million, respectively:

$$X = BP$$

where X is the total amount of private financial protection required, in dollars; B is the "base" amount of protection required, that is, $150 times the authorized thermal kilowatts; and P is a "population factor." The population factor P is based on the term

$$\sum_i \frac{N_i}{D_i^2}$$

where N_i is the total population of the ith "minor civil division" enclosed in or intersected by a circle of radius R miles, with R in miles being numerically equivalent to the square root of the reactor power in thermal megawatts and the population being that determined in the 1960 or later census; D_i is the distance in miles (to the nearest mile but not less than 1 mile) from the reactor to the geographic center of the same minor civil division as that used in establishing the value of N_i. The established values of P are listed below in relation to ranges of values of $\sum_i N_i/D_i^2$:

$\sum_i N_i/D_i^2$	P
0–1000	1.0
1000–3000	1.1
3000–5000	1.2
5000–7000	1.3
7000–9000	1.4
9000–10,000	1.5

Some questions have arisen regarding the "rightness" and adequacy of the details of these specifications. For example, should a factor be included to account for the inventories of the more biologically hazardous fission products (e.g., I^{131} and Sr^{90})? Should there be a factor included to account for differences in degree of certainty of containment of radioactivity? Should a specification be included to correlate the risks with factors that associate population concentrations with prevailing wind directions? What is the proper basis for deciding the amount of private protection required, beyond which government indemnification is guaranteed? Should there be a maximum amount of potential government liability? If so, should it be a constant (presently $500 million), or should it also vary by some formula? Is the distance to the center of a geographic area a good criterion for risk to the population? Is the use of an inverse-square distance function equally valid for all installations?

An anomaly may arise from the requirement of $60 million private protection for reactors authorized for 100 *electrical* megawatts while the requirements for lower electrical capacities are based on *thermal* megawatts. It would be possible for a lower thermal power station with a high efficiency to be required to furnish more protection than a higher thermal power station with a lower efficiency. Although differences in insurance rates might adjust the cost inequities, there is no assurance that they would do so.

The Commission has indicated its willingness and desire to hear objections and proposals. It has pointed out that the regulations are subject to continuous study for improvement. The two largest interested associations of insurance companies, Nuclear Energy Liability Insurance Association (NELIA) and Mutual Atomic Energy Liability Underwriters (MAELU), have stated their objections to some parts of the proposed amendment.[45] Their comments are based on the premises that the government would be entering into direct competition with private business where private business has ample capacity and willingness to do the job and that the formula underrates the "logical" amount of insurance required for power reactors of less than 100-Mw(e) capacity, especially on the lower end of that range. Some alternate formulas have been proposed by the associations, along with specific curves of insurance requirements versus reactor power and population distribution. The Commission has met with representatives of the associations to hear these objections and proposals.

A commentary on financial protection provisions in the Atomic Energy Act, federal regulations and proposed amendments to federal regulations, and proposed insurance contracts has been published by the Atomic Industrial Forum, Inc.[46] The committee pointed out several questions of coverage and of interpretation involved in adjudication of assumed nuclear protection insurance contracts (proposed by NELIA and MAELU). Their analysis showed "gaps" in coverage caused by (1) possible cancellation or suppression of insurance, (2) incomplete coverage of property damage, (3) limitations on contract liability coverage, (4) coverage during transportation of radioactive materials to a facility, (5) possible delays in reinstatement of full coverage after an incident, and (6) possible deficiency of coverage for the aggregate of several reactor sites where the $60 million limit might be exceeded. The analysis also brought out serious questions of coverage of the licensee in (1) risks at the reactor site during the period prior to license, (2) questions of interpretation of "holding the government harmless" under section 53e(8) of the Act, (3) risks of liability for damage to the reactor and other "on-site" property, (4) questions of definitions of "nuclear incident" and "location," and (5) conflict of duty to the insurer and to the government.

In summary, it seems clear that the aims of the new proposal are, in general, in the "right direction": to emphasize the causes of risk and to improve the correlation of financial protection requirements with the risk involved. The questions and objections remaining may be divided into three types: (1) technical, (2) legal completeness and interpretation, and (3) philosophy of government in determining its proper relations with private enterprise in protecting the general public. It is also clear that the

proposal is intended to be a step in improvement rather than a complete solution. (L. A. Mann)

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Fuel-element Failures

In the design of any reactor system, the problems associated with, and the consequences of, fuel-element failures require considerable detailed analysis. There are three basic approaches to the problem: design the fuel elements so that they will not fail, incorporate a detection system so that any failed element may be readily located and replaced, or design the coolant system so that the fission products released from a reasonable number of faulty elements may be tolerated. The first approach is practically impossible and economically prohibitive in a reactor system where neutron economy is important and structural materials must be minimized. The second method is the one which today appears to be uniquely available to some types of gas-cooled reactors. The third approach is, of necessity, the trend in most reactor systems striving for competitive power production.

A number of reports[1–5] have appeared recently on the subject of fuel-element failures and the resulting fission-product activity in and external to the reactor system. One of the reports[1] presents an analytical study of the effects of fission-product leakage in the N.S. *Savannah* reactor. In this study differential equations are derived for determining the buildup of activity in the primary coolant and in the purification and waste-disposal system. The escape-rate coefficients for the leakage of fission products from the fuel elements, as used in the equations, were based on the third PWR test run at Chalk River.[6] One hundred and twenty-six nuclides of 25 elements are included in the calculations of the activities.

The report on a second analytical study[2] discusses the effects of a fuel-cladding failure on the activity in the coolant of the proposed 45.5-Mw organic moderated reactor. In this system the fuel elements are enriched metallic uranium plates metallically bonded to the aluminum cladding. After a cladding failure of such an element, three processes for fission-product release to the coolant are postulated: (1) direct recoil of the fission fragment to the coolant, (2) release by corrosion processes of fission products which have built up in the fuel, and (3) fission of uranium which has been corroded from the fuel and is circulating in the coolant.

The rate of corrosion of the uranium fuel (and fission products present in the uranium) by the coolant was assumed to remain constant with time and to be 9.65×10^{-11} cm/sec for an average temperature of 500°F. This corrosion rate was based on experiments conducted at 350°F with the organic coolant open to the atmosphere. This condition permitted oxygen and water to dissolve in the organic material and thereby to contact and corrode the uranium. More recent experiments performed at 700°F in a sealed autoclave gave a corrosion rate of 6.8×10^{-11} cm/sec for as-cast natural uranium. Both these experiments were carried out with the coolant quiescent and do not take into account any erosion of the fuel.

The results of the study[2] indicated that the maximum dose rate at the outer surface of a 16-in. schedule-40 coolant pipe would be 0.25 mr/hr per square millimeter of fuel surface exposed. Since the normal activity to be expected from the coolant impurities, the corrosion, and the recoil products is about 250 mr/hr, rather large cladding-failure areas could be tolerated before the activity from the failure became appreciable.

Additional escape-rate coefficients for a defective Zircaloy-2-clad UO_2 fuel pellet have been obtained in a test at Chalk River.[3] In this latest test, the heat rating on the element was about twice that for which data were previously reported,[7,8] but the results were approximately the same as those for the lower heat flux tests. The test was terminated after about two weeks because of a sudden unexplained burst of activity during a reactor trip-startup cycle. The data were compared with those reported by Bettis Atomic Power Laboratory (BAPL), as indicated in Table II-1.

Table II-1 COMPARISON OF BAPL AND X-2-q ESCAPE-RATE COEFFICIENTS

	BAPL	X-2-q
Rate of heating, $\int k(\theta)\, d\theta$, watts/cm	24	48
UO_2 density, g/cm^3	10.2	10.2
Grain growth of UO_2	Trace	Extensive
Escape-rate coefficient, sec^{-1}:		
Kr^{88}	1×10^{-8}–2×10^{-8}	6.4×10^{-8}
Cs^{138}	2×10^{-8}–4×10^{-8}	3.5×10^{-8}
I^{131}	10^{-9}–10^{-8}	1×10^{-8}*
I^{133}	1×10^{-8}–5×10^{-8}	6×10^{-9}*

*Values obtained just prior to activity burst which terminated the test.

An Alco Products, Inc., report[4] gives the results of an experimental program for determining the identity, level, and origin of fission products and other short-lived nuclides in the primary coolant of the Army Package Power Reactor, SM-1 (APPR-1). The report describes the various approaches taken to establish the source of the activity, as well as the experimental techniques and mode of reactor operation used during the period of the test. The conclusion is reached that the major source of activity is fuel-element defects, with some contribution from surface contamination. Although the conclusion is probably correct, three of the four arguments presented to justify the conclusion appear to be incorrect. The three arguments in question are based on showing that the release of the fission products is by a diffusion mechanism rather than by recoil from surface contamination. At the maximum temperatures (600°F) reached in the APPR fuel plates, the release of fission products even from exposed UO_2 would be predominantly by recoil rather than by diffusion.[9]

The results of operating the BORAX-IV reactor with 22 of its 69 fuel elements defective have been reported by Robertson and Hall.[5] These special tests were conducted over a two-day period after it became apparent that a new core loading for the reactor contained a number of defective elements. During reactor operation, measurements were made of the activity levels of the steam-plant equipment and the quantities of fission gases (Xe^{138} and Kr^{88}) released through the air ejector. The reactor water and condensed steam before the turbine and the condensed steam after the turbine (hot-well condensate) were analyzed for fission products, and the area downwind from the reactor was monitored for contamination. Special metal coupons were also placed in the turbine to determine the activity deposited in this area during the test run.

It was found that the fission products released were primarily fission gases and that products without fission-gas precursors were not released to any significant extent. It was concluded from the tests that it is possible to operate a boiling-water reactor for long periods of time with many cladding defects if the system is free from leaks and there is sufficient holdup volume in the line to the stack for a fission-gas delay time of at least 1 hr.

Further information[10] has been obtained on the release of fission products during the recent Sodium Reactor Experiment (SRE) incident.[11] The average burnup of the damaged fuel elements was 725 Mwd/metric ton, with a maximum of 1136 and a minimum of 299. It is estimated that the release of xenon, krypton, and iodine was 300, 250, and 350 curies, respectively, or about 0.3 per cent of the activity of the elements which had cladding failures. A summary of the data and general conclusions of the investigation to date are presented in Table II-2.

Table II-2 SUMMARY OF FISSION-PRODUCT CONTAMINATION IN SRE COOLANT FOLLOWING FUEL-ELEMENT FAILURES

Isotope	Coolant activity,* μc per gram of Na	Total coolant activity,* curies	Total reactor activity,* curies	Fraction of total reactor activity released to coolant†
Cs^{137}	1.26	2.77×10^1	8.70×10^3	3.18×10^{-3}
Cs^{134}	0.02	4×10^{-1}	2×10^2‡	2×10^{-3}‡
Sr^{89}	20.0	4.44×10^2	1.60×10^5	2.78×10^{-3}
Sr^{90}	0.97	2.14×10^1	8.15×10^3	2.63×10^{-3}
I^{131}	0.74	1.63×10^1	1.68×10^4	0.97×10^{-3}
Ce^{141}	4.38	9.65×10^1	1.27×10^5	0.76×10^{-3}
Ce^{144}	5.18	1.41×10^2	1.69×10^5	0.67×10^{-3}
Ba^{140}–La^{140}	1.65 (each)	3.63×10^1 (each)	5.61×10^4 (each)	0.65×10^{-3}
Zr^{95} + Nb^{95}	13.9 (sum)	3.06×10^2 (sum)	5.53×10^5 (sum)	0.55×10^{-3}
Ru^{103}	0.95	2.09×10^1	7.52×10^4	0.28×10^{-3}

*As of reactor shutdown on July 26, 1959.

†Multiply values in this column by 3.5 (ratio of total number of fuel elements to number that failed) to adjust fraction released to that released from only those fuel elements which had cladding failures.

‡Estimated values for activity from neutron captures by Cs^{133}.

Data from samples taken from the reactor cover gas were much more limited, and extrapolation back to reactor shutdown is of questionable accuracy because of the more mobile nature of the cover gas, which is used for flushing and is subject to pressure changes. However, the data for the two isotopes found in the cover gas are presented in Table II-3.

Table II-3 SUMMARY OF FISSION-PRODUCT CONTAMINATION IN SRE COVER GAS FOLLOWING FUEL-ELEMENT FAILURES

Isotope	Cover-gas activity,* $\mu c/cm^3$	Total cover-gas activity,* curies	Total reactor activity,* curies	Fraction of total reactor inventory released to cover gas
Xe^{133}	7.4	4.7×10^{1}	5.08×10^{4}	0.92×10^{-3}
Kr^{85}	0.016	2.0×10^{-1}†	1.1×10^{3}	0.18×10^{-3}

*As of reactor shutdown July 26, 1959.

†Includes factor of 2 for probable losses due to gas flushing, etc.

The investigation of the eventual fate of the fission products released from the fuel elements is still in progress, and little quantitative information is presently available. However, the following general observations have been made.

Although the iodine content of the coolant was somewhat lower than expected in consideration of its relative volatility, no indication of its presence was ever found in the cover-gas samples. This is in agreement with all the previous gas samples taken at the SRE, some of which were designed to greatly concentrate any iodine present and thus distinguish it from the usual contamination (e.g., Xe^{133}). A possible explanation of the low value for iodine contamination of the sodium could be a greater rate of removal of iodine (by deposition, etc.) from the sodium relative to the rate of removal of the other fission products present. The sodium was circulated through the system for about one week after reactor shutdown before the first sodium sample was taken because of the Na^{24} activity present.

Large amounts of fission-product deposition from the sodium occurred during the continuing circulation. Succeeding sodium samples showed decreases in fission-product contamination far in excess of reductions that could be attributed to radioactive decay. The mechanisms of the deposition have not yet been well established, but it is known that (1) the carbonaceous material present in the sodium had a specific fission-product activity much greater than that of the sodium, perhaps by an average factor of at least 10^3; (2) the cold trap in the sodium system was concentrating the fission-product activity in some manner, but whether this occurred by the collection of the carbon particles or the fission products directly or both is not known; and (3) drainage of the sodium from the reactor system did not result in a decrease (in fact, an increase was noted) in the radiation levels in the pipe galleries, and thus it appears that there was appreciable deposition in the pipes.

The low values for the apparent xenon and krypton gas releases may be due, in part, to the capture of these elements in the void space provided at the top of the fuel rods. Since most of the fuel elements parted at or below the midpoint, all the previously accumulated gas may not have escaped. Such an effect would be much more important for Kr^{85} than for Xe^{133}, since more than 93 per cent of the Kr^{85} inventory was produced prior to the last run. (G. Samuels)

Chemical Reactions of Reactor Materials

Chemical reactions that proceed slowly over long time intervals during reactor operation markedly influence reactor design, expected reactor lifetime, and the economics of power production. They may, of course, be the cause of reactor incidents and may, under some conditions, magnify the severity of such an incident. However, the reactions that are rapid and energetic and those which contribute products that are explosive or especially hazardous are the primary concern in nuclear safety considerations. Some recent studies of several classes of chemical reactions that may contribute to nuclear incidents are described here.

Oxidation of Reactor Metals

The study of oxidation of active metals used as fuels, coolants, claddings, etc., is receiving attention at several laboratories. Oxidation reactions are obviously important in the safety of nuclear devices since, if they can occur, they will greatly magnify the consequences of a reactor malfunction.

Oxidation of Alkali Metals. The potential hazard of a sodium fire is a problem common to all sodium-cooled reactors. A study of the design criteria for a containment vessel for the Enrico Fermi reactor, undertaken by Atomic Power Development Associates, Inc., as part of its over-all study of sodium-cooled reactors, has been reported.[12] The report states that all components of the reactor system which contain radioactive sodium are located in an inert atmosphere in the bottom portion of a 72-ft-diameter containment vessel underneath a 5-ft-thick concrete operating floor. Access ports to any sodium-containing components are closed with steel-lined concrete plugs. If any of the pipes or components containing sodium were to leak, the sodium would collect in a pool at the bottom of the containment vessel under the inert atmosphere.

For the study of the potential hazards, it was assumed that a major leak in the sodium circuit had occurred and that the seals had failed between upper and lower compartments of the containment vessel so that air from the upper compartment was available to burn the sodium. The results of two experiments in which about 50 lb of preheated sodium was burned in air showed that 5 lb of sodium per hour per square foot of exposed sodium surface would be a reasonable value for the initial rate of combustion of the sodium from the reactor; this figure is in reasonably good agreement with a value estimated by British investigators. From these experiments,[12] Garelis concludes that combustion takes place on the sodium surface and that the rate, accordingly, is controlled by the concentration and velocity of the oxygen molecules. This leads to an expression such as

$$B = T^{1/2}\, k\, n_{O_2}$$

where B is the combustion rate, k is a constant, n_{O_2} is the concentration of oxygen molecules, and T is the temperature. With the use of the experimental results and time-dependent heat balances on the reactor-containment building and its contents, the maximum gas temperature and pressure expected from the postulated incident were obtained. These values are given as 1115°F and 26 psig and are stated to be well within those tolerable for the proposed containment vessel.

Calculations were also made by Garelis[12] to show that, for the case of a constant-volume adiabatic reaction of sodium and air, a pressure of 9.75 atm (129 psig) and a temperature of 3554°F would be realized. This pressure value may be compared with a value of 6.4 atm (79 psig) obtained at Argonne[13] when sodium was explosively injected into a 3- by 10-ft cylinder of air in amounts corresponding to 3.14 moles of sodium per mole of oxygen. It was reported to be very difficult to achieve the degree of dispersion necessary to obtain such pressures. The Argonne report,[13] accordingly, concludes that in an actual reactor installation it would be impossible to obtain complete dispersion of sodium and air and that the pressures which could be developed would not approach either those predicted by calculation or those realizable from small-scale experiments.

A report[14] from MSA Research Corporation contains a brief discussion of hazards associated with liquid-metal handling and disposal. Although this brief presentation includes little, if any, new material, it serves as a complete index of previous MSAR documents in which experimental studies are discussed in some detail.

Oxidation of Uranium. Experimental studies[15] at Argonne have established that a number of surface contaminants, including aluminum dust, ethyl alcohol, cutting oil, silicon carbide, water, and the corrosion products from exposure to water, retard or have no effect on the rate of propagation of burning along uranium foils in air. However, previous etching of the uranium foil in concentrated nitric acid significantly increases the burning propagation rate. Burning propagation rates for uranium foils of 5 mils (0.13 mm) in thickness have been shown to vary with foil width. In air, 2- and 4-mm-wide strips burn at rates that are within the range of average deviation of the burning rates of 3-mm-wide strips, but strips 1 mm in width burn significantly faster. In oxygen the pattern is not so clear; 1-mm-wide specimens burn at the fastest rates, 4-mm-wide specimens are next, and no definite trend is evident.

The studies at Argonne have included examinations of the effect of light and of the heating method on the oxidation behavior of uranium at low temperatures. Some of these experiments were conducted with the uranium specimens on the hot stage of a microscope so that the course of the oxidation could be examined. Neither marked acceleration of the initial oxidation nor any unusual appearance of the specimen surface

could be observed when the specimen was irradiated with the tungsten-filament lamp during oxidation. A similar experiment in which the specimen was nearly completely shielded from room light yielded similar results.

Experiments were also conducted in which the specimen was held at 200°C. In these experiments, induction heating, heating by external resistance windings, and heating by a combination of the two methods were compared. There was some evidence that induction heating may have affected the oxidation rate, but the differences noted were probably due to the marked differences in the temperature of the oxygen gas in these experiments. Resistance heating, which heated the oxygen to temperatures as high as those of the specimen, resulted in definite first- and second-stage oxidation behavior. Induction heating, which left the oxygen gas unheated, gave no real evidence of first-stage oxidation. Heating by a combination of the two techniques resulted in behavior that was intermediate in character. The oxidation of the uranium was more rapid in cool oxygen than in oxygen heated to the specimen temperature. Observations of nodule growth on the surface of the uranium oxide in these experiments, along with the absence of significant effects of additives to the metal and the dependence of the oxidation rate on oxygen pressure, suggest that the rate-controlling step in oxidation of uranium occurs at the oxygen-oxide interface. Observations of electropolished specimens and specimens roughened by abrasion with various grades of silicon carbide are consistent with this hypothesis.

Oxidation of Zirconium and Its Alloys. When zirconium foil strips burn in air, they maintain their shape, and a narrow oxidation front precedes the ignition front. When similar samples burn in oxygen, they are completely converted to the oxide, and globules of molten metal and oxide drip from the foil strip as the burning progresses. In measurements of burning propagation rates of zirconium alloys, foil strips 0.13 mm thick and 0.6 mm wide were studied in air, and strips 0.13 mm thick and 3.0 mm wide were studied in oxygen. A copper-zirconium alloy showed the slowest burning rate in air, and a titanium-zirconium alloy showed the fastest. Propagation rates were about 20 times more rapid in oxygen, with the rate for a nickel-zirconium alloy being slower than that for pure zirconium; titanium-zirconium alloys burned most rapidly.

When zirconium or zirconium alloy specimens burn in air, the burning area behind the ignition front maintains a relatively constant length. The length of this "burning zone" differs for each alloy, but a plot of this length against propagation rate yields a reasonably straight line for all alloys observed. A sharp increase in propagation rate with decreasing ignition temperature is observed for ignition temperatures above 700°C in air or oxygen; for lower ignition temperatures, propagation rates increase less markedly. The data suggest a limiting propagation rate of about 1 cm/sec in air and about 17 cm/sec in oxygen.

Unlike the behavior of uranium under oxidation, pure zirconium burns in oxygen at a rate which depends on the cross-sectional area. However, for cross-sectional areas above 0.3 mm^2, the effect is relatively small.

In shielded ignition experiments at Argonne, ignition temperatures of pure zirconium in moist air and moist oxygen have been shown to be higher than those in the dry gases; water vapor is not, therefore, expected to increase the rate of burning of bulk zirconium. Adding liquid water to burning zirconium is, on the other hand, extremely hazardous.

A small effect of surface preparation on the ignition temperature of zirconium foil (0.17 mm thick) in oxygen has been observed. Chemically polished specimens, for example, showed ignition temperatures of 842°C, while as-received specimens ignited at 833°C.

Metal-Water Reactions

Some experiments on reactions of molten zirconium and uranium metal with water have been reported.[15,16] In these experiments, metal wires, usually 60 mils in diameter, were rapidly melted and dispersed in a water-filled cell by a surge current from a bank of condensers. The energy input to the wire indicated the reaction temperature; the transient pressure was a measure of the reaction rate; the extent of reaction was obtained by determining the amount of hydrogen produced; and the particle size of the residue indicated the surface area exposed for reaction. These studies showed that fast, explosive pressure rises occur when zirconium is allowed to react at initial temperatures above 2600°C. Methods for measuring the particle sizes of the

residual powder have been developed. The data have not yet been completely analyzed, but it is tentatively concluded that a change must occur in the mechanism of the reaction in the vicinity of 2600°C; perhaps the ZrO_2 melts at 2700°C.

The MSA Research Corporation report[14] referred to above also contains a brief summary of previously reported studies of (1) metal-water reactions, (2) air-borne activity resulting from small leaks and from catastrophic leaks in contaminated water systems, and (3) escape of fission products upon meltdown of Zircaloy-clad zirconium-uranium alloy fuel subassemblies in steam atmospheres. The report lists previous MSAR reports that present the findings in greater detail.

Other Chemical Hazards

Reaction of Organic Moderators and Coolants. Experimental evaluations of some possible hazards of a biphenyl type of coolant are also briefly described in the MSA Research Corporation report.[14] Two attempts to simulate a steam-generator failure in a biphenyl system were made. No evidence of chemical reaction or of any unusual effects was observed when biphenyl was forced at 800°F and 500 psig into water at 200°F and 300 psig or when water at 200°F and 500 psig was forced into biphenyl at 800°F and 300 psig. Heating of a NaK-biphenyl mixture at atmospheric pressure under a nitrogen cover produced no evidence of chemical reaction. The biphenyl vaporized at 500 to 540°F without incident, and no further changes were apparent on heating to 800°F.

Reaction of BeO with Water. An expression describing the corrosion rate of a beryllia surface in moist flowing air has been developed and treated by experiment in studies at Atomics International.[17,18] Data were obtained which support the hypothesis that the rate-controlling step in the corrosion process is the diffusion of the gaseous beryllium hydroxide species through the laminar boundary layer at the beryllia surface. The corrosion rate r, in mils per year, is given by

$$r = 6.85 \times 10^{10} \frac{D}{T_f t}(K p_w - p_b)$$

where D is the diffusion coefficient of beryllium hydroxide through air, in square feet per second; T_f is the average temperature of the boundary layer, in degrees Fahrenheit; t is the boundary-layer thickness, in feet; K is the equilibrium constant; and p_w and p_b are the partial pressures, in atmospheres, of the water vapor and the beryllium hydroxide, respectively, in the bulk gas stream. Data from 20 experiments are presented. In two experiments the BeO showed no loss in weight; in 15 of the remaining 18 experiments, the corrosion rates of the BeO specimens agree to within a factor of 2 with values calculated from the above equation. McKisson states[17] that the corrosion reaction will not limit the feasibility of operation of an open-cycle air-cooled beryllia-moderated power reactor at beryllia surface temperatures below 1560°F.

Disposal of Gases from Fuel-element Dissolution

The processes used to dissolve spent reactor fuel elements for reprocessing and recovery of the fissionable and fertile materials present a variety of hazards that include release of volatile and intensely radioactive fission products from the dissolving material. Removal of this gaseous activity, as well as any particulate activity that may be carried in the stream, must be accomplished before the off-gas may be discarded or released, and, moreover, the problem is often complicated by the presence of reactive or even explosive major components of the off-gas mixture.

A brief discussion of potential hazards to be expected from the off-gas system from a power reactor fuels reprocessing pilot plant has been presented by Holmes.[20] In the pilot plant the head-end process consists of a sulfuric acid dejacketing step followed by nitric acid dissolution of the fuel meat. Xenon, krypton, and iodine will be liberated in up to 25 scfm of hydrogen in the dejacketing step and in much smaller quantities of gases containing oxides of nitrogen during the final dissolving operation.

In one of the processes described, the rare gases would be scrubbed from the hydrogen in liquid nitrogen and concentrated to 5 to 10 per cent in a distillation column. The hydrogen would then be burned in a submerged combustion system. Although such a scheme appears to be feasible, it would require all the safety features necessary for both compressed-hydrogen plants and air-liquefaction plants. The problem is

further complicated by the requirement that the off-gas system also handle the oxides of nitrogen from the HNO_3 dissolving step.

In an alternate process the hydrogen would be removed by submerged combustion or by catalytic combustion before recovery of the rare gases from nitrogen either by partial condensation or, perhaps, by adsorption on low-temperature charcoal. The document describes and contains references to safety practices in liquid hydrogen and air separations plants. No definitive solution to the off-gas problem is presented. (W. R. Grimes)

Comparison of Maximum Credible Accidents Postulated for U. S. Power Reactors

The operation of any nuclear facility in which large amounts of radioactive materials are present represents a potential hazard to the safety of the general public. In order to assure that such facilities are designed and operated in the safest possible manner, numerous safety features are provided in the design of the reactor which either prevent the occurrence of possible accidents or minimize the consequences of accidents. Thus reactor fuses, spare coolant pumps, emergency power systems, etc., are employed to prevent the occurrence of an accident which could lead to the release of radioactivity. On the other hand, devices such as cooling sprays, ventilation filters, and containment vessels are used to minimize the consequences of an accident.

Evaluation of the need for any of the safety features in a particular system is currently determined largely on the basis of subjective judgment by the reactor designer and by the members of two AEC groups, the Hazards Evaluation Branch of the Division of Licensing and Regulation and the Advisory Committee on Reactor Safeguards. The hazards reports submitted by reactor designers to the AEC groups for review discuss the probable accidents and define and evaluate the maximum credible accident and its consequences.

Table II-4 gives design data pertaining to the maximum credible accidents that have been postulated and described in the hazards reports for existing and proposed U. S. power reactors. The maximum credible accidents assumed by the various reactor designers, as well as the consequences of the assumed accidents, are discussed here. Only the type of accident and the amount of activity released are considered. Subsequent Reviews will present discussions of other aspects of the maximum credible accident, such as containment requirements and dose rates to personnel.

Pressurized-water Reactors

Army Package Power Reactor (SM-1). The maximum credible accident for the SM-1 (formerly the APPR-1)[21] occurs when the reactor is running constantly at the maximum power and, through an extremely unlikely sequence of failures, the temperature of the water in the primary and secondary systems rises to the saturation temperature, whereupon a rupture occurs and releases the stored energy into the vapor container. If the reactor core melts during the incident, a maximum of 10^8 curies is released.

Shippingport Atomic Power Station (PWR). The maximum credible accident for the PWR results from the rupture of the primary coolant system.[22,23] Like several of the other pressurized-water reactors, the PWR has a safety injection system to prevent core meltdown or to minimize damage to the core (see Table III-1, page 39 of this Review). If this system does not function properly at the time of the loss-of-water accident, the seed elements begin to melt after 2 min and are completely melted in 23.1 min. Blanket meltdown starts after 13 min but is not complete for 203.8 min. For breaks in the coolant system below the level of the active core and larger than 6 in. in diameter, the safety injection system will only limit the extent of core meltdown. In the hazards analyses for the PWR, the effects of various amounts of activity release were investigated, the largest activity release considered being release of 100 per cent of the volatiles and 1 per cent of strontium. Therefore this amount of activity has been included for the maximum credible accident.

Consolidated Edison Thorium Reactor (CETR). The maximum credible accident for the CETR[24,25] results from the loss of water following rupture of a 24-in.-diameter primary loop pipe. Three pumps in the seal-water and primary-water makeup system are available to

supply water to the primary system. However, for a rupture in the system equal to or greater than the area of a 4-in.-diameter pipe, the leakage rate is always greater than the makeup rate, and the system is drained to the elevation of the rupture. Therefore, at the time of the maximum credible accident, the lack of adequate cooling will cause the cladding on the fuel to melt, and the UO_2 fuel may collect at the bottom of the pressure vessel, where large amounts of fission products will be released.

Yankee Nuclear Power Station. The maximum credible accident for the Yankee reactor[26] results from the rupture of a 20-in.-diameter line in the primary system. It is assumed that the safety injection system functions to prevent fuel meltdown, and therefore the maximum credible accident does not result in any appreciable release of fission products. In the hazards report the consequences of a hypothetical accident which results from primary system rupture and failure of the safety injection system are analyzed. As a result of this accident, the fuel cladding begins to melt about 155 sec after the system rupture and is completely melted in 22 min. The estimate of fission-product release from core meltdown is based on experiments with fuel elements of the PWR type.

N.S. Savannah Power Plant. The maximum credible accident for the N.S. *Savannah* reactor[27,28] results from the rupture of the primary coolant piping. When the reactor does not receive adequate cooling, the fission-product decay heat will cause the core temperature to rise. Eventually all fuel elements will fail because of the cladding failures, and finally there will be complete meltdown of the core. No safety cooling system is provided to prevent core meltdown or to minimize the damage to the core.

Saxton Reactor Facility. The maximum credible accident for the Saxton reactor[29] results from the rupture of the primary system coupled with failure of the safety coolant injection system. In the analyses of the loss-of-coolant accident, it is assumed that the pressure vessel does not rupture but that the rupture of a 4-in.-diameter line will cause partial (exact extent not defined) meltdown of the core. It is estimated that the cladding on the fuel elements will begin to melt 4 min after the system rupture with no emergency cooling. The fission-product release data are based on experimental melting of fuel elements of the SM-1 and PWR types in a steam atmosphere.

Boiling-water Reactors

Vallecitos Boiling Water Reactor (VBWR). The maximum credible accident for the VBWR[30] will be a boiler-explosion type of accident, 100 per cent static failure without an accompanying nuclear excursion or a metal-water reaction. From the time of severance of a 10-in.-diameter water outlet line, the operator will have 8 min to supply coolant from a shutdown cooling system before the zirconium cladding melts and 14 min before melting of UO_2 begins. The maximum credible release of activity will ensue if this cooling is not provided, and complete melting of fuel will occur. Nevertheless, it is claimed that no credible combination of circumstances would result in substantial release of fission products to the enclosure.

Dresden Nuclear Power Station. The maximum credible accident for the Dresden nuclear power station[31] will be initiated by the instantaneous complete severance of one of the bottom inlet lines to the reactor while the reactor is in the "hot stand-by" condition. After the accident, it is assumed that nearly all the water that is not separated by a solid barrier will be released within 1 min from the reactor and all auxiliaries. Melting of the cladding and a $Zr\text{-}H_2O$ reaction will start in 10 min, fission-product volatilization will start in 15 min, the UO_2 will begin to melt in 16 min, and fuel melting will be complete in 40 hr. The reactor has no spray cooling provisions.

The "worst reasonable accident" will occur if all the pressurized hot water in the reactor and primary system is suddenly released and a nuclear excursion follows with enough energy to melt all the nuclear fuel. In addition, it is assumed that 25 per cent of the zirconium cladding will react rapidly with the water.

Elk River Reactor. The maximum credible accident for the Elk River reactor[32] is a loss of primary coolant through complete rupture below the core of a 16-in.-diameter forced-circulation line which results in drainage of all water in less than 10 sec. Continued operation of the spray cooling system is assumed, but the worst

Table II-4 DATA RELATIVE TO POSTULATED MAXIMUM CREDIBLE ACCIDENTS

Reactor type and designation	Thermal power, Mw	Core power density, watts/cm^3	Fuel-element materials		Maximum fuel-element operating temp., °F		Coolant pressure, psi	Fission products released from fuel elements that failed as result of maximum credible accident, %		
			Fuel body	Cladding	Cladding surface	Central		Noble gases	Halogens	Other
Pressurized-light-water Reactors										
SM-1	10	71.7	UO_2 in stainless-steel matrix with B_4C	Stainless steel	554	742	1200	100	100	100
PWR	230	75 in seed; 24.6 in blanket	Seed, uranium–Zircaloy-2 alloy; blanket, UO_2	Zircaloy-2 on both seed and blanket elements		740 in seed; 3800 in blanket	2500	100	100	1
CETR	585	76	UO_2–ThO_2	Stainless steel	605	4350	1500	25	25	6.25
YAEC	485	74	UO_2	Stainless steel	663	4330	2000	20	20	Sr, 20
NMSR	69	21.1	UO_2	Stainless steel	623	3794	1750	100	100	Rb, 100; Cs, 100; Sr, 10; Ba, 10; Te, Se, Sb, Eu, Mo, Te, and Sm, 1; Y, La, Ce, Pr, Nd, Pm, Zr, Nb, Ru, and Rh, 0.1
Saxton	20	34.3	UO_2	Stainless steel			2000	50	25	0.5
Boiling-water Reactors										
VBWR	20 (up to 40 Mw 5% of time)	50	2.3% enriched UO_2	Stainless steel or zirconium	590	4600	1165	100	100	30
Dresden	630	30	1.4% (av.) enriched UO_2	Zircaloy-2	675	4700, maximum hot-spot temperature about 5000°F (for new fuel)	1015	100	100	10
Elk River	58	39.6	ThO_2–UO_2 pellets; U is 93.5% enriched	Type 304 stainless steel 0.020 in. thick	566	3420	936.5	100	100	Sr, 5
Pathfinder (includes internal superheater	203	96	1.8% enriched UO_2 (core);	X-8001 Al alloy tubing	1330 in super-	4800 in core	615	100	50	1

			U is 93.5% enriched	0.020 in. thick						
Pathfinder (includes internal superheater in two-region single core)	203	96	1.8% enriched UO_2 (core); 93% enriched UO_2 (superheater)	X-8001 Al alloy tubing in core; stainless steel on cermet elements in superheater	1330 in superheater	4800 in core	615	100	50	1
Humboldt Bay	163	25.5	1.9% enriched UO_2	Zircaloy-2	700	4570	1070	100	25	Volatile solids (Ru, Se, Te, Cs), 15; other solids, 0.3
Big Rock Point	240	45	3.9% enriched UO_2	Stainless steel	700	4655 (max)	1050	100	25	2
BONUS (includes nuclear superheater)	50	33 in core; 12 in superheater	1.85% enriched UO_2 plus natural U plus 3.5% enriched UO_2 in superheater region	Zircaloy-2	1140 in superheater; 640 in core	4000 in superheater; 4400 in core	915	100	100	30
Other Types										
HRT (fluid fuel)	10	17.3					2000	100	100	100
Fermi (sodium-cooled fast reactor	300	857	Uranium-molybdenum alloy	Zirconium	1055	1316	20–120			
EGCR (helium cooled)	84 in reactor; 10 in loops	1.87 in reactor	UO_2	Stainless steel	1575	3300	300 in reactor	100	100	10
Peach Bottom (helium cooled)	Case III, 115	Case III, 8.57	Cases I–III, graphite + UC + ThC mixture	Case I, metal sheath; Case II, internal metal sheath; Case III, graphite	1530 in Cases I and II; 2400 in Case III		300	Case I, 100; Case II, 0; Case III, 100	100 0 0	0 0 0
HNPF (graphite moderated, sodium cooled)	240	4.2	Uranium-molybdenum alloy	Stainless steel	855	1260	1–6 in H_2O	0	0	0
Piqua (organic moderated)	46	18.75	Uranium-molybdenum alloy	Aluminum	750	880	120	100	100	100
CVTR (pressurized D_2O tube reactor)	60	7.3	UO_2	Zircaloy-2	584	4700	1500	20	60	Sr, 1
FWCNG (CO_2 cooled, D_2O moderated, pressure tube)	153	4.91	UO_2	Stainless steel	1500	3800	500	20	20	20

likely spray distribution results in the melting of the cladding of 21.2 per cent of the fuel pins. The extent of melting was calculated by assuming that all cooling water that enters the core flows down the inside of the Zircaloy shrouds and directly cools only the shrouds and the fuel pins adjacent to the shrouds. For the other fuel pins, it is assumed that all heat will be transferred by radiation only to the outer pins.

Pathfinder Atomic Power Plant. The maximum credible accident for the Pathfinder reactor[33] is complete severance of a reactor coolant-recirculation line. It is assumed that the water and flash steam will be released directly to the free volume of the reactor building within 5 to 10 sec. The aluminum-clad fuel elements within the core will then melt and release fission products. This will be followed by the melting of the stainless-steel-clad fuel elements in the superheater region of the core within 15 min. Although molten aluminum cladding will flow or fall to the water below, no metal-water reactions are expected. The reactor has no emergency spray cooling provisions.

Humboldt Bay Power Plant. The maximum credible operating accident for the Humboldt Bay reactor[34,35] results from the near instantaneous rupture of a main steam or feed-water pipe. The backup core-cooling system is designed to function so that the fission products not released coincident with the primary system break would be essentially retained at their source in the fuel. In the event that the rupture results in drainage of the core vessel, the spray from either feed line is adequate to prevent core meltdown. The mechanical action of the escaping steam and water may damage fuel elements, and the sudden decrease in external cladding pressure may cause some fuel elements with weak cladding to burst. Therefore approximately 1 per cent fission-product release to the dry well may result, the release being predominantly gaseous fission products. If the backup core-cooling system fails, decay heat will cause the cladding to rupture about 1 min after system rupture, and one-third of the cladding will melt in the first hour. Melting of the fuel will begin 15 min after system rupture, and one-fifth of the fuel will melt in the first hour; all the fuel will melt in 24 hr.

Big Rock Point Nuclear Power Plant. The maximum credible accident for the Big Rock Point reactor[36] will be caused by a main steam or feed-water pipe break with the reactor in "hot stand-by" condition. It is assumed that the core spray cooling system will be unavailable, in which case core melting will start in 15 min and be completed in 40 hr. However, the core spray cooling system is designed to prevent significant cladding or fuel melting following a rupture in the primary system, including one in which all coolant is lost within 2 sec.

Boiling Nuclear Superheater Power Station (BONUS). The maximum credible accident for BONUS[37] will result from the complete severing of a 16-in.-diameter line at the bottom of the pressure vessel coupled with failure of the emergency core spray cooling system. As a result of such an accident, the water in the pressure vessel will discharge in about 4 sec, and the cladding on the fuel in the superheater region of the reactor will start to melt 30 sec after the break; however, the average time to melt the cladding will be 16 min. The UO_2 will start to melt in 6.7 min, whereas the average UO_2 element will melt in 80 min. If the core spray cooling system is operative, it may be shown that there is sufficient water supply for the core spray cooling system to remove decay heat continuously for 8 hr, even though 20 per cent of the water is evaporated.

Other Reactor Types

Homogeneous Reactor Test (HRT). The maximum credible accident for the HRT[38] results from the rupture of the primary system and consequent release of activity into the containment cell. Since the HRT is a fluid-fueled reactor, all the fission products will, at the time of the system rupture, escape from the system. The activity level of the circulating fuel is somewhat reduced by the continuous utilization of a cleanup system and also by deposition on system surfaces.

Enrico Fermi Reactor. The maximum credible accident for the Fermi reactor[39,40] has not been defined (as of December 1959), but various analyses of accidents indicate that, in the unlikely event of a primary system rupture, the core will not melt. The primary system layout is such as to provide a pool of sodium to remove decay heat at all times. Piping below the elevation of the center line of the exit sodium lines is double walled, and syphon breaks are installed

in the coolant lines. Therefore, since no single failure will result in lowering the level of sodium in the core, meltdown of the fuel is not expected.

Experimental Gas-cooled Reactor (EGCR). The maximum credible accident for the EGCR[41] results from the double-ended failure of the primary system piping. One of the blowers is assumed to be capable of operating and provides sufficient heat removal to prevent extensive core damage. It is calculated that one fuel element will fail and release about 200 curies of activity because of the degradation of cladding strength and the internal pressure in the fuel element from the buildup of the fission gases xenon and krypton.

The maximum credible accident for the reactor, if operation of the experimental loops is included, results from the double-ended failure of the primary system followed by buckling of the experimental tubes in the reactor core. Such an accident could release large amounts of activity from the experimental loops since adequate cooling to the experimental assemblies cannot be assumed.

The activity release from the fuel in the core is based on escape-rate data which are extremely sensitive to the temperature at which the fuel operates and also to the method of fabrication of the fuel. Likewise the failure rate is sensitive to the diffusion coefficient of xenon in UO_2 and to the temperature of the fuel both during operation and at the time of the maximum credible accident. Uncertainties in temperatures or diffusion coefficients make it difficult to predict the amount of activity release per fuel-element failure or the number of fuel-element failures to be expected. Inasmuch as the activity release from the loops would be so much greater than the activity release from the fuel in the reactor, the uncertainties as to the release from the fuel are not a significant factor in determining the consequences of the maximum credible accident.

Peach Bottom Atomic Power Station. In the hazards analyses for the Peach Bottom reactor,[42] various accidents were investigated for the three types of fuel indicated in Table II-4. In all cases the maximum credible accident results from the rupture of the primary coolant system, and the heat removal is therefore limited to that of the afterheat removal system. At the time of the maximum credible accident, the fuel elements with metallic sheaths will rupture from fission-gas pressure. The fuel elements with internal metal sheaths, as well as those with graphite cladding, are expected to retain their integrity and to allow only small amounts of fission products to be released. It is assumed for the metallic-clad fuel that all the fission products which are volatile at core operating temperatures are available for release during the maximum credible accident and that 80 per cent of all the fission products, except the noble gases and halides, will condense in passing from the fuel element to the containment vessel.

Hallam Nuclear Power Facility (HNPF). The maximum credible accident for the HNPF[43] results from the failure of a control-rod thimble in the region of the helium atmosphere in the reactor. As a result of the failure, the reactor atmosphere mixes with the thimble atmosphere, and the fission gases which enter the thimble become a source of direct gamma-ray doses. The activity in the coolant during normal operation is postulated to result mainly from fuel-element failures and sodium activation. For the analyses, it is assumed that the cladding fails on 1 per cent of the fuel elements, that 1 per cent of the fission-gas activity is released to the rod void spaces, and that 20 per cent of the fission products which exist in the fuel are gaseous. It is also assumed that 60 per cent of the activity released from the fuel is iodine and that the iodine is absorbed in the coolant. Based on these assumptions, the noble-gas activity available to mix with the gas in the control-rod thimble at the time of the maximum credible accident is 2×10^4 curies.

Piqua Reactor. The maximum credible accident for the Piqua reactor[44] results from the continuous and uncontrolled withdrawal of control rods from the core during startup of the reactor. The incident is terminated by void formation and meltdown of 50 per cent of the six center fuel elements and release of 7.5 per cent of the core fission products. However, the integrity of the coolant system is assumed, and therefore no appreciable amounts of activity are released outside the coolant system. In the case of loss of the coolant by pipe rupture, the decay-heat-removal system prevents core meltdown.

Carolinas-Virginia Tube Reactor (CVTR). The maximum credible accident for the CVTR[45] results from the severance of a 10-in.-diameter pipe and the simultaneous failure of the emergency injection system. The injection system is fed into both the inlet and outlet headers to assure that coolant is provided to the core regardless of where the rupture occurs. In order to prevent core meltdown, the injection system must operate within 8 sec from the time of the primary system rupture. For the maximum credible accident, since the injection system is assumed to fail, the fuel cladding melts and large amounts of activity are released.

Florida West Coast Nuclear Power Plant (FWCNG). The maximum credible accident for the FWCNG reactor[46] is not specified in the hazards report. However, the loss of the system pressure has been investigated for two cases: (1) a case in which the only activity released results from leakage of fission products from pinhole defects in 0.3 per cent of the fuel elements and (2) a case in which fission-product release results from cladding failures of 20 per cent of the fuel elements. For the first case, the activity release is based on escape-rate data from the X-1 loop at the NRX reactor at Chalk River. For the second case, it is assumed that 20 per cent of the fission products are released from the failed elements to give a total fission-product release of 4 per cent of the total fission products in the core.

Discussion

In comparing the maximum credible accidents for the various reactors, it is not surprising to find that the various reactor designers do not all have the same concept of the maximum credible accident. The lack of operating experience for the various reactor types leads to great uncertainty as to the credibility of an assumed accident, the sequence of events leading to an accident, and the probable consequences of the accident. For a particular reactor type, it therefore happens that one reactor designer assumes a credible accident that does not appear to another reactor designer to be credible. Inasmuch as most reactor plants have containment vessels capable of containing the released activity from a complete core meltdown, the design is often based on a "hypothetical accident" rather than the maximum credible accident.

In addition to the uncertainty as to the credibility of accidents, there is an equal uncertainty by reactor designers as to the consequences of a given maximum credible or hypothetical accident. Although it is always safe to assume that the worst result of an accident is the release of all the fission products from the reactor system (and therefore safe to design the system to accommodate the release), such an assumption unnecessarily penalizes the reactor economically and, in the long run, would probably eliminate the system from competing on an economic basis with other reactors or other power sources.

Several of the factors which influence analyses of (1) the types of accident which lead to the maximum credible accident and (2) the consequences of the maximum credible accident are discussed below. There are, of course, many factors other than those discussed here, and therefore the reader should refer to the specific hazards report for additional and more detailed information.

Size and Location of Rupture. With few exceptions, reactor designers define the maximum credible accident as that resulting from a loss-of-coolant accident. A rupture in the primary cooling system is proposed as the mechanism by which the coolant is lost. In none of the postulated maximum credible accidents has it been considered that the loss-of-coolant accident could occur in the reactor pressure vessel itself; rather, the rupture is assumed to occur in the reactor piping. Even here there is considerable variation from one report to another as to the size and location of the break in the piping, although the most common set of conditions assumes a double-ended pipe failure at a point which would effect the greatest loss of primary coolant from the reactor core. Pressure-vessel failures are apparently excluded because of confidence in the integrity of these vessels when built according to the ASME code. On the other hand, piping, including expansion joints, valves, etc., is generally subjected to a wider range of thermal and mechanical stress. The size of the system rupture will determine the time sequence of the resulting meltdown, whereas the location of the rupture will determine whether the coolant level (in the case of liquid-cooled reactors) uncovers the core and whether the emergency cooling system, which must connect to the primary cooling system at some point, will be effective.

System Pressure. The pressure of the primary coolant system will influence the maximum credible accident in two ways. First, the high-pressure systems are more subject to rupture, and, following the rupture, the coolant of the high-pressure system will be expelled at a more rapid rate that depends on the size of the rupture. The pressure in the primary system will also influence fuel-element failure, especially when the fuel has unbonded cladding. During operation, fuel elements build up internal pressure which is resisted by the coolant external pressure. When the coolant system ruptures, the pressure differential across the fuel-element cladding may increase rapidly and cause the fuel elements to burst. In low-pressure systems, such as the Hallam and Fermi reactors, the rupture of the coolant system is improbable, and in the case of rupture the core would not melt.

Type of Fuel Element. At the time of the maximum credible accident, activity may be released from the fuel elements in the reactor core if adequate cooling is not provided. The amount of activity that is released will depend, among other things, on the type of fuel in the reactor. In some reactors the maximum credible accident will not result in fuel failure, and in other reactors the fuel may fail by bursting from fission-gas pressure buildup or may melt either partially or completely.

The meltdown of fuel elements will result in the release of large amounts of the volatile fission products. Meltdown experiments have been performed on various types of fuel,[47] but none of the experiments have included fuel elements of the type generally proposed for use in power reactors. Therefore it is not surprising to see a wide range in the assumed values of fission-product release from melted fuel elements.

Excessive pressure differentials across the cladding of fuel elements as a consequence of the maximum credible accident will lead to bursting of the cladding of unbonded elements. This type of failure may lead to substantially less fission-product release than the meltdown and will depend on the fuel-element power density, the temperature during normal operation, the temperature transient at the time of the accident, and the extent of fuel burnup. Estimates of the fraction of the activity released by this type of fuel failure may be based on escape-rate data or diffusion coefficients. However, the experimental information available with which to make the calculations is so limited as to cast serious doubts on the results.

Emergency Cooling Systems. Various reactors depend upon some type of emergency cooling to prevent or minimize the amount of fuel failure in the event of loss of coolant as a result of pipe ruptures. In general, the water-cooled reactors depend upon an emergency ejection system to prevent core meltdown, whereas the gas-cooled reactors depend upon operation of the blowers to provide adequate cooling in the event of a system rupture. The adequacy of any cooling system will, of course, depend on several factors, and the inclusion of an emergency system may not necessarily prevent core meltdown unless certain requirements are met. Probably of greatest importance is the requirement that the emergency system be entirely independent of the primary coolant system so that the failure of the latter will not prevent the emergency system from operating as required. At the time of a coolant-system rupture, the emergency system must be activated within a relatively short period of time to prevent core meltdown, the time depending on such factors as the size of the rupture and the power density in the core. Cooling systems for the various reactors are compared in Sec. III of this Review.

Conclusions

This comparison of the maximum credible accidents postulated for the various U. S. power reactors leads to the following conclusions:

1. The maximum credible accident for the pressurized-water, boiling-water, and gas-cooled reactors is caused by the rupture of the primary coolant system. Failure to provide adequate emergency cooling following coolant-system rupture will lead to extensive fuel-element failures.

2. The pressurized- and boiling-water reactors operate at high power densities, and therefore, at the time of the maximum credible accident, inadequate cooling will result in partial or complete meltdown of the core. Thus these systems may be expected to require emergency cooling systems.

3. The gas-cooled reactors considered operate at relatively low power densities, and therefore, at the time of the maximum credible

accident, core meltdown is prevented by circulating air through the core with one of the blowers of the primary loop. Damage to the fuel is limited to the bursting of fuel elements.

4. The probability of failure of the primary system of the sodium-cooled or organic-cooled reactors is appreciably less than for the gas-cooled or water-cooled reactors because of lower system pressures. In the event of a rupture, the core does not melt and fuel-element failure should not occur.

5. The release of fission products from the fuel elements at the time of the maximum credible accident is subject to question for all the postulated accidents. The lack of experimental information concerning the behavior of the UO_2 fuel used in most of these reactors under the conditions postulated makes the consequences of the maximum credible accident questionable.

6. The credibility of an assumed accident is subject to uncertainty in that operating experience is insufficient for accurate analysis, and the probability of a single or a multiple failure is equally uncertain. Therefore the general practice has been to provide some form of containment to give the greatest assurance of safety. The absolute need for containment will, however, not be known until additional operating experience is gained and more extensive research and development are carried out.

(C. G. Bell and H. N. Culver)

Reactor Accidents

Fast, self-limiting power excursions have been studied, both experimentally and theoretically, for a number of different reactors. In the first issue of this Review,[48] several of these studies were discussed which dealt primarily, although not exclusively, with water-moderated reactors in which the limiting mechanism involved transfer of heat to the moderator and expulsion of some of the moderator from the core. In the second issue of this Review, several kinetics problems of light-water-moderated reactors were discussed.[49] Attention is given here to two cases in which the energy generated in a power burst was largely retained within the fuel elements, and the excursion was terminated principally by the negative temperature coefficient of reactivity of the fuel. Another reactivity effect discussed here is that due to the loss of the moderator. This effect has been examined experimentally at the Savannah River Laboratory for a number of special lattices.

In water-moderated and -cooled reactors, it is often found that the most serious accident situation results from a rupture of the primary coolant circuit during full-power operation. Analyses of instances involving such an accident in the Plutonium Recycle Test Reactor and in a pressurized-water reactor are discussed.

Termination of Reactor Transients by Temperature Coefficients

The transient behavior of TRIGA, a light-water-cooled reactor having homogeneous fuel-moderator elements composed of uranium and zirconium hydride, has been studied by Stone et al.[50] The core of TRIGA consists of up to ninety-one 1.5-in.-diameter cylinders of uranium (20 per cent enriched) and zirconium hydrided to approximately 1 hydrogen atom per zirconium atom. The cylinders have a fueled length of 14 in. and are clad in 0.030-in.-thick aluminum. Step reactivity changes are accomplished with the use of a boron carbide rod which may be removed from the core in less than 0.1 sec.

In initial experiments it was found that uniform heating of the core produced slight changes in reactivity, varying from $+1.0 \times 10^{-5}$ $\delta k/°C$ at 10°C to -0.8×10^{-5} $\delta k/°C$ at 60°C. If only the fuel-moderator elements are heated, however, as is approximated in operation at appreciable power levels, a large loss of reactivity results. In a series of steady-state experiments at various power levels up to 1.4 Mw, it was found, after correcting for the slight coolant temperature rise, that the fuel-temperature coefficient of reactivity varied from -1.2×10^{-4} $\delta k/°C$ at 0°C to -0.9×10^{-4} $\delta k/°C$ at 400°C. These results may be applied to the prediction of transient behavior in a prompt burst.

In a series of burst experiments, with reactivity steps from 0.6 to 1.6 per cent δk, the reactor achieved asymptotic periods down to 10 msec and peak powers up to 250 Mw. The maximum fuel-temperature rise reached in the 1.6 per cent δk burst was 360°C, corresponding to a total energy release of 16 Mw-sec (10 Mw-sec in the prompt burst).

For a step reactivity insertion of 1.6 per cent, the power level, P, during the transient is given very well by the Fuchs model,

$$\frac{1}{P}\frac{dP}{dt} = \frac{1}{\tau} - b\int_0^t P(t)\,dt$$

where τ is the asymptotic period in seconds and b is the energy shutdown coefficient, which was found for TRIGA to be 2×10^{-5}/watt-sec^2. From this expression, it follows that the maximum power in watts is

$$P_{max} = 2.5 \times \frac{10^4}{\tau^2}$$

and the energy in the burst in watt-seconds is

$$E = \frac{10^5}{\tau}$$

These expressions appear to be in good agreement with the observations for reactivity insertions greater than one dollar. For smaller reactivity insertions, the agreement is not so good, but it would not be expected to be, since the model neglects delayed-neutron and heat-transfer effects. This successful application of the Fuchs model is somewhat at odds with SPERT experience,[51] but it is evident that the neglect of heat-transfer effects was more appropriate in this case than it would have been in the SPERT case. Calculations based upon a detailed space-independent heat- and neutron-flux model of the reactor satisfactorily account for the reactor behavior following the prompt burst, but they are discussed only briefly in the paper.[50]

The prompt negative temperature coefficient of the hydride fuel-moderator elements derives principally from neutron-spectrum effects associated with the bound hydrogen. Doppler broadening of U^{238} resonances and prompt expansion of the fuel elements (which displaces water from the core) together contribute about 20 per cent of the observed reactivity coefficient. During the prompt excursions, very little heat is transferred to the coolant.

Another very interesting case is that of TREAT, a reactor designed to melt test fuel elements in a central test hole during fast transients, while causing no damage to the reactor core itself. This is accomplished by having a much higher ratio of power density to heat capacity in the test element than in the reactor core. On the other hand, the reactor fuel temperature must rise enough to terminate the power excursion through the negative fuel-temperature coefficient. These considerations are discussed in a paper by Iskenderian,[52] which deals with most of the reactor physics aspects of the TREAT reactor. The core is an essentially homogeneous mixture of graphite and U^{235}, with a carbon-to-U^{235} atom ratio of 10,000. This ratio is a compromise in that heavier loadings of uranium produce a shorter prompt-neutron lifetime and hence a shorter reactor period for a given reactivity insertion, whereas lighter loadings produce a lower ratio of power density to heat capacity (or, alternatively, a higher neutron flux in the test element). Iskenderian estimates the temperature coefficient to be $-2.5 \times 10^{-4}/^\circ$C; it derives mainly from an increase in the thermal-neutron diffusion length and the non-$1/v$ behavior of the U^{235} cross section. Progress of the power excursion for a reactivity insertion of considerably more than one dollar is calculated with this temperature coefficient and the assumption that all the heat generated remains in the fuel-moderator elements of the core. The expression employed for the reactivity is

$$k_{ex} = At - c\int_0^t n(t)\,dt$$

where A is a constant for $t \leq t_0$, the duration of the ramp insertion, and $A = k_0/t$ for $t > t_0$ and k_0 = terminal value of ramp. For the case of a very rapid insertion, the reactivity expression becomes

$$k_{ex} = k_0 - c\int_0^t n(t)\,dt$$

which, as shown below, is identical with the Fuchs model. In order to show the identity with the Fuchs model, let

$$\frac{\dot{\phi}}{\phi} = \frac{1}{\tau}$$

Since

$$\tau = \frac{l}{k_{ex} - \beta}$$

$$\frac{\dot{\phi}}{\phi} = \frac{k_{ex} - \beta}{l}$$

Substituting for k_{ex} yields

$$\frac{\dot{\phi}}{\phi} = \frac{1}{\tau_0} - b\int_0^t \phi(t)\,dt$$

where $b = c/l$ and

$$\tau_0 = \frac{l}{k_0 - \beta}$$

is the asymptotic period of the reactor after insertion of the reactivity but before appreciable power generation.

With this analysis, Iskenderian predicts, for a reactivity insertion of 0.02, a minimum period of 30 to 40 msec, a peak flux of 10^{17} neutrons/(cm^2)(sec), and a total energy release of 580 Mw-sec. (The core dimensions are 4 by 5 by 5 ft.) The maximum temperature rise in the core is estimated to be not greater than 300°C. Since approximately as much energy is generated after the peak as before, it follows that the reactivity inserted may be, at most, half the product of the temperature coefficient times the allowable temperature rise. For example, for an average temperature rise of 235°C, the product is

$$235^\circ\text{C} \times 2.5 \times \frac{10^{-4}}{^\circ\text{C}} = 0.06$$

and the maximum reactivity insertion is 0.03. These conditions are calculated to be more than ample to melt specimens of natural-uranium metal in the test hole.

Reactivity Effects of Loss of Moderator

Moderator-level control has often been suggested for D_2O-moderated power reactors and is incorporated in the design of the Canadian NPD-2 reactor both for regulation and for emergency shutdown. The reactivity effect of moderator loss depends upon the particular lattice and fuel element involved and requires separate investigation for each case of interest. At the Savannah River Laboratory this effect has been studied experimentally[53] for several special-purpose lattices in which, owing to the unique design of the fuel elements, it was expected that loss of moderator would be accompanied by an increase in the infinite multiplication factor, k_∞. The need for an experimental approach to this problem arises principally from the difficulty of calculating the migration area in lattices which are highly anisotropic and which contain large void spaces.

In a series of exponential and critical experiments, it was shown that complete loss of moderator, while retaining coolant (H_2O, D_2O, or a mixture) in the process tubes, usually resulted in a substantial increase in k_∞. In most cases the loss of moderator resulted in a change in the spatial distribution of neutrons which increased the relative rate of neutron absorption in the fissionable material. In lattices containing control rods, it was found that the reactivity worth of the rods was lower by a factor of 5 to 10 in the unmoderated lattices than in the moderated lattices. In all cases, however, it was shown that increased migration area and neutron-streaming effects more than offset the increased k_∞. Consequently, the k_{eff} was decreased by loss of moderator in every case studied, including that of a large reactor with geometrical buckling equal to 0.00013.

Temperature Excursions Following Loss-of-coolant Accident

A loss-of-coolant accident is serious for several reasons. First, the large amount of energy stored in the pressurized coolant would be released. The possible energy release is so large, in fact, that in most cases the design of the containment vessel is governed by this energy. Second, the loss of coolant would usually result in melting and possible vaporization of the fuel elements, with attendant release of fission products. Even if the fission-product activity were confined to the containment vessel, the damage to the plant might be great. Third, the molten fuel elements might react chemically with coolant remaining in the pressure vessel and thus increase both the temperature and the pressure of the contained atmosphere. It is clearly worthwhile, if at all possible, to prevent melting of the fuel following such an accident, and injection of water from an emergency supply into the reactor is a commonly considered means of accomplishing this.

Extensive analyses are required to ensure that the fuel elements can be kept below their melting points for any assumed type of failure of the primary coolant system. Such analyses have been performed at Battelle Memorial Institute for the Plutonium Recycle Test Reactor (PRTR),[54] a heavy-water-moderated and -cooled reactor fueled mainly with Zircaloy-clad UO_2, but also containing some plutonium-aluminum alloy elements. The calculations dealt almost entirely with the time-dependent temperature structure in the reactor and were based on water- and steam-flow calculations performed at Hanford.[55] The reactor was assumed to be

scrammed following a low-pressure signal, and only fission-product decay heat was considered to be important after the first second or so following initiation of the scram. Ruptures of the coolant circuit at several different points were analyzed, and each postulated failure was found to lead to a somewhat different pattern of coolant supply and fuel temperatures after the failure. Three distinct phases were considered in each case. During the first phase, which consisted of blowdown of the coolant until the pressure was equalized between the coolant system and the containment vessel, the fuel would be cooled sometimes by water and sometimes by steam. Therefore the inventories and flow rates of the coolant and heat-transfer coefficients were computed as functions of time. The second phase considered was the adiabatic phase after the coolant pressure reached equilibrium, during which there was no cooling of the elements; and the third phase was the emergency water-injection phase, which would start when the pressure in the system fell below the injection pump head, i.e., about 100 psig.

In all cases examined, it was found that the fuel would cool during the first phase, with some of the element temperatures falling essentially to the coolant temperature. In the adiabatic phase, of course, the elements would heat up again, and if emergency water injection did not start within about 3 min, melting of the plutonium-aluminum alloy elements would occur. The onset of melting of the Zircaloy-clad UO_2 elements would, in general, take 10 to 15 min. The study suggests that conditions could possibly arise, particularly for certain small ruptures, in which the pressure would remain above the specified injection pump head too long after portions of the elements were uncovered and that therefore the emergency water-injection phase would be delayed too long. Consequently, an injection system capable of operating against a higher system pressure was recommended. These results highlight the necessity of a detailed analysis to ensure that emergency measures will actually function as intended. The calculations involved in this study were carried out on an IBM-653 computer, and the program for the calculations is discussed in detail in an appendix to the report.[54]

The process of coolant expulsion from the primary coolant system of a pressurized-water reactor following a rupture has been discussed in detail by Harris.[56] The paper deals mainly with expansion of the saturated coolant from the pressurized volume after the pressure has fallen to the saturation vapor pressure of the coolant at its initial temperature. This initial pressure decrease is shown to occur very rapidly. Mass-balance and heat-balance equations are derived on the assumption of a succession of equilibrium thermodynamic states. The extreme cases of complete steam-water separation and of no separation are considered. For small ruptures and low rates of efflux, the first extreme should be applicable. For larger ruptures, the actual case will fall somewhere between the two extremes, which serve to bracket the problem.

No details of coolant flow, such as were employed by Lemmon et al.[54] to compute heat-transfer coefficients, are provided for in Harris' formulation, except that the coolant inventory in the second-pass region of a two-pass core is described by a separate equation. However, within the assumption that the rate of efflux is limited by the orifice formed by the rupture (isostatic system), these details can be added when the geometry of the system and the location of the rupture are specified. The effect of pressure buildup in the containment volume, which is relevant only in the last stages of the blowdown, is omitted from Harris' analysis. He does assume that emergency water can be injected at a pressure equal to the initial saturation vapor pressure of the coolant (i.e., early in the blowdown process), and his equations provide for such injection. As Harris points out, the mode of coolant expansion, with or without steam separation, both determines and depends upon the flow rate and may be quite uncertain in a given case. However, the behavior of the system can be bracketed and a conservative solution adopted. (A. M. Perry)

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Emergency and Shutdown Cooling Systems of U. S. Power Reactors

All reactor designs provide for some sort of cooling method (or methods) to remove heat from the core following a credible accident, as well as following a shutdown from normal operation. Such cooling may be obtained from an emergency cooling system installed specifically for the accident situation and/or from the normal shutdown cooling provisions. The cooling methods required will depend upon answers to questions such as: (1) is the reactor inherently safe from the possibility of meltdown of fuel elements and spread of radioactive material following the maximum credible accident, (2) how fast and how reliably can the reactor be shut down following the maximum credible accident, (3) if the primary coolant is lost following the maximum credible accident, what action must be taken to prevent meltdown of the core, and (4) is the primary system configuration such that a rupture could be isolated and the remaining operable system components be used for heat removal? The emergency and shutdown cooling systems for various reactors are discussed separately here because each reactor will have either one or both systems.

Some of the basic design data of the shutdown cooling systems described in the preliminary hazards reports of 19 different reactors[1–19] are listed in Table III-1. The reactors listed are either operating, being constructed, or being designed. It should be pointed out that changes may have been made in the specifications of the shutdown cooling systems of the various reactors since the hazards reports were issued and that such changes are not reflected in this comparison.

Emergency Cooling Systems

An emergency cooling system, as referred to here, is composed of auxiliary devices, in addition to those of the shutdown cooling system, which are provided to protect the reactor core following a credible accident. Therefore the emergency cooling system must be based on considerations of the nature and magnitude of such accidents. For example, such a system is frequently provided if core meltdown could occur. In the review of 19 different reactors, it was found that six of the designs included emergency cooling systems.

The Yankee pressurized-water reactor,[1] the Shippingport pressurized-water reactor,[2] the Army package power pressurized-water reactor,[3,20] the Saxton pressurized-water reactor,[5] the Elk River boiling-water reactor,[11] and the Carolinas-Virginia pressure-tube reactor[17] have safety injection systems that protect the core upon loss of primary system pressure as a result of a rupture of the primary piping. The injection water is supplied through spray rings or nozzles or to the cold leg of the coolant loop at the reactor pressure-vessel inlet. The flow of injection water must be such that the core is covered with water until any danger of meltdown has passed. It should be noted that this condition can exist only if the failure in the primary system does not cause partial or complete drainage of the core. If the core becomes partially uncovered, core meltdown may occur even though the continued operation of the emergency cooling system may effect some cooling. Safety injection systems are designed to operate until the rupture is isolated and the primary system or additional auxiliary equipment is operable for removal of heat from the core. If it is not possible to isolate the rupture, the flow of water must continue until the core can be safely allowed to drain of all coolant.

Emergency cooling systems are generally not placed into operation until needed. These systems are therefore subject to the uncertainties usually associated with systems which are normally inoperative. They must depend upon instrumentation to detect the need, and this instrumentation must instigate action, i.e., open valves and/or start a pump. Furthermore, both the instrumentation and the coupling component

Table III-1 EMERGENCY AND SHUTDOWN COOLING SYSTEMS

Reactor name	Reactor type*	Primary coolant	Thermal power, Mw	Core power density, watts/cm^3	Emergency cooling provided	Relation of shutdown cooling system to primary cooling system
Yankee Nuclear Power Station	PWR	H_2O	485	74	Yes	Independent
Shippingport Atomic Power Station	PWR	H_2O	230	75 in seed, 24.6 in blanket	Yes	Dependent
Army Package Power Reactor (SM-1)	PWR	H_2O	10	71.7	Yes	Dependent
Indian Point Nuclear Power Station	PWR	H_2O	585	76	No	Independent
Saxton Reactor Facility	PWR	H_2O	20	34.3	Yes	Dependent
N.S. *Savannah*	PWR	H_2O	69	21.1	No	Independent
Dresden Nuclear Power Station	BWR	H_2O	630	30	No	Independent
Pathfinder Atomic Power Plant	BWR	H_2O	203	96	No	Independent
Humboldt Bay Power Plant	BWR	H_2O	163	25.5	No	Independent
Vallecitos Reactor	BWR	H_2O	20	50	No	Independent
Elk River Reactor	BWR	H_2O	58	39.6	Yes	Independent
Experimental Gas-cooled Reactor	GCR	He	84	1.87	No	Dependent
Florida West Coast Nuclear Power Plant	GCR	CO_2	153	4.91	No	Dependent
Hallam Nuclear Power Facility	SGR	Na	240	4.2	No	Dependent
Sodium Reactor Experiment	SGR	Na	21	4.4	No	Dependent
Enrico Fermi Atomic Power Plant	SGR	Na	300	857	No	Dependent
Carolinas-Virginia Nuclear Power Plant	PTR	D_2O	60	7.3	Yes	Independent
Homogeneous Reactor Test	AHR	UO_2SO_4-D_2O	10	17.3	No	Independent
Piqua Reactor	OMR	Organic	46	18.75	No	Independent

*PWR, pressurized-water reactor; BWR, boiling-water reactor; GCR, gas-cooled reactor; SGR, sodium graphite reactor; PTR, pressure-tube reactor; AHR, aqueous homogeneous reactor; OMR, organic-moderated reactor.

(line and valve) must be operative even in the event of the maximum credible accident.

A reactor that does not have an emergency cooling system must rely on inherent safety or system configuration to provide adequate protection. A reactor that has four primary coolant loops is inherently safer than one with only one loop, inasmuch as a rupture in one of four loops could possibly be isolated and the remaining three operated at reduced load. Reliance on such a multiplicity of loops and components may be misleading, however, since a single rupture inside the block valves to the loops could result in the loss of water in the entire system.

The emergency cooling system of the reactor must be considered part of the safety system; consequently the system must be designed for equivalent reliability. This means that the necessary power supplies should be in multiple units, coolant supplies should be adequate, and instruments and components should be reliable and should be tested regularly. The control of the emergency cooling system should be such that a malfunction of any instrument or component would not result in damage to the primary system.

Shutdown Cooling Systems

Shutdown cooling systems are provided for the long-term removal of reactor decay heat. In general, shutdown cooling systems differ from emergency cooling systems in that they are not capable of supplying coolant to the system at a rapid rate to compensate for loss via rupture. The reactors reviewed here have one of two general types of shutdown cooling system, i.e., an independent or a dependent system.

An independent shutdown cooling system is one that does not depend on any primary system components except the reactor pressure vessel

and piping to the isolation valves. Additional pumps, heat exchangers, water supplies, etc., are installed to remove the decay heat after the primary system temperature has been reduced to a safe level, if it is desired to isolate the remainder of the primary system for inspection or maintenance.

A dependent shutdown cooling system is one that must rely on virtually all the primary system components but which has supplementary devices for increasing system reliability for removal of all heat following any accident, as well as during and following a normal shutdown. Such devices may be pony motors for driving main circulators, separate circulators in a bypass around the main circulators, steam-driven feed-water pumps, and stand-by water storage.

Independent Shutdown Cooling Systems. Eleven of the 19 reactors considered are provided with auxiliary shutdown cooling systems that are virtually independent of the primary reactor cooling systems. With the exception of the organic-moderated reactor, these 11 reactors are water-cooled and include the pressurized-water reactors, the boiling-water reactors, and the Homogeneous Reactor Test. As in the case of the emergency cooling system, operation of the shutdown cooling system is usually dependent on action taken upon information provided by instrumentation in the primary system. However, in contrast, the time constants associated with the operation of the shutdown cooling systems may be much greater.

The Yankee pressurized-water reactor,[1] the Elk River boiling-water reactor,[11] and the Carolinas-Virginia pressure-tube reactor[17] shutdown cooling systems consist of one or more auxiliary heat exchangers, circulating pumps, and a secondary water supply to remove the decay heat generated in the reactor core. The system is placed in operation when the system temperature and pressure have been reduced to some predetermined level. The system continues long-term removal of heat to permit safe maintenance and inspection.

The Pathfinder boiling-water reactor[8] shutdown cooling system removes decay heat by natural circulation to an emergency condenser located 20 ft above reactor level. The shell side of the condenser is filled by gravity from a storage tank. The condensed steam is returned by gravity to the reactor primary system.

The Dresden boiling-water reactor,[7] the Humboldt Bay boiling-water reactor,[9] and the Vallecitos boiling-water reactor[10] include a special heat exchanger or condenser which is supplied with cooling water from a gravity storage tank for the removal of decay heat following a normal shutdown. The Indian Point pressurized-water reactor,[4] the N. S. *Savannah* pressurized-water reactor,[6] and the Piqua organic-moderated reactor[19] provide for the forced circulation of the primary coolant through external heat exchangers from which it is returned to the primary system for long-term decay-heat removal.

The Homogeneous Reactor Test[18] shutdown cooling consists of an auxiliary cooling-water system external to the primary system components which removes decay heat from vital components, such as the reactor pressure vessel, circulating pumps, letdown heat exchangers, entrainment separators, and space coolers. This system operates continuously and is thus not dependent on the reliability of instrument action or upon components which must be operated or started at the time of shutdown.

Dependent Shutdown Cooling Systems. Eight of the 19 reactors considered have shutdown cooling provisions which are dependent upon the integrity of most, if not all, of the primary system components. The very nature of these systems implies that these reactors have less need of shutdown cooling or are not as subject to as serious failures as the reactors in which independent shutdown cooling systems are provided. It is of interest to note that the reactors in this category include the two gas-cooled reactors considered, the three sodium-cooled reactors, and three of the pressurized-water reactors.

The experimental gas-cooled reactor[12] and the Florida West Coast Nuclear Group gas-cooled reactor[13] provide auxiliary blowers, or auxiliary blower drive motors, to remove the decay heat after a normal shutdown following a loss of main coolant circulation. The primary system heat exchangers must be in operation.

The Hallam sodium graphite reactor[14] and the Sodium Reactor Experiment[15] do not have auxiliary shutdown cooling systems. They are dependent on natural-convection cooling of the sodium through an arrangement that prevents the pressure vessel from being completely drained of coolant and exposing the core.

The Shippingport pressurized-water reactor,[2] the Army package power pressurized-water re-

actor,[3] the Saxton pressurized-water reactor,[5] and the Enrico Fermi sodium graphite (fast) reactor[16] rely entirely on thermal-convection cooling for removal of decay heat. The heat pump is provided by water which is supplied to the secondary system from an elevated makeup-storage tank.

General Requirements

On the basis of the considerations presented here, it would be extremely difficult to arrive at an unqualified generalization as to the emergency and/or shutdown cooling requirements of any given reactor. Further, it is not apparent that the designs of the various reactor types are consistent with regard to these features.

All reactor systems provide for shutdown cooling, of course, and it is not particularly surprising to find that for some reactor types these requirements are less stringent than for others. In particular, the sodium-cooled reactors are capable of removing afterheat merely by thermal convection of the coolant because of the excellent heat-transfer properties of sodium. The two gas-cooled reactors, EGCR and FWCNG, likewise do not require independent shutdown cooling systems, but for a vastly different reason, namely, the low power density of the system. It is apparent, however, that the attainment of higher power densities and/or higher operating temperatures in future gas-cooled reactors may require firmer shutdown cooling provisions.

The pressurized-water reactors show the greatest variation in emergency and shutdown cooling requirements. Some have emergency cooling systems whereas others do not, and some have dependent shutdown cooling systems whereas others have independent cooling systems. There is no apparent correlation with any reactor design features or operating conditions. Inasmuch as all these reactors are installed within containment vessels, there has perhaps not been the incentive to define the need more specifically.

All boiling-water reactors have independent shutdown cooling systems, but only one has an emergency cooling system. The relatively high power densities in these reactors apparently require good shutdown cooling. Again, until such time as the monetary value of emergency cooling may be evaluated in terms of containment requirements, the necessity and hence the provision of emergency cooling will probably remain the option of the designer.

Neither the emergency cooling systems nor the so-called "independent" shutdown cooling systems can be completely independent of the primary coolant system, since there must be a common tie point. Failure at this junction could prevent the operation of the special cooling system. In addition, both the shutdown and the emergency cooling systems depend upon instrumentation to detect the need for cooling and upon the operation of such components as may be required to effect the cooling. Although not employed by any of the reactors discussed above except the HRT, the most reliable type of independent cooling system is one which is normally operating so that, in the event of need, its operability is not subject to the failure of sensing instrumentation or the failure of cooling-system components to operate as required.

(R. M. Pierce)

Preoperational Defect Tests of Clad Fuel Elements

The detection of defective fuel elements before use in a reactor is, of course, essential to the safety of the system. Experience has shown that defective elements can cause fission-product contamination of the primary coolant system and may lead to serious failures in the reactor core. Means for detecting fuel elements that have failed were discussed in the previous issue of this Review.[21] The discussion here is concerned with the detection of clad-fuel-element defects before the elements are installed in a reactor.

The nondestructive evaluation of the fuel element may include inspection of the materials and components before fabrication as well as the completed element or assembly. The purpose is to detect any defect which will permit a leak of fission products or to disclose flaws which are likely to develop into leaks. Among the inspection methods which may be used are visual inspection, radiography, dye penetrants, thermal effects, ultrasonics, electrical resistance, eddy currents, gas leak tests, and the reaction of the fuel with a substance forced through the leak. The test methods adopted depend, of course, on the fuel-element configuration. There are two general types of clad fuel elements. In one type, the fuel material is encapsulated in

such a fashion that the outer capsule or cladding is not bonded to the fuel material. Thus, even though the cladding may be in contact with the fuel material, there exists a void volume within the cladding which is interconnected. In the other type of fuel element, the cladding is metallurgically bonded to the fuel material so that there is no void volume within the fuel element, except that which may exist within the pores of the fuel matrix.

Unfortunately, no single test will prove conclusively the existence or absence of a leak or defect. The presence of a flaw indication as the result of a proper inspection usually demonstrates the existence of an actual defect; however, failure to obtain a flaw indication does not necessarily mean that no flaw is present. This latter condition may arise because the specific inspection method being used is incapable of detecting the type of defect present or because of improper use of the proper technique. In most instances a number of nondestructive tests must be used with great care to assure that the fuel elements accepted are of the desired quality. The validity of the nondestructive methods used must be established by destructive tests and, finally, by the use of the fuel elements in a reactor. The destructive tests usually consist of extensive metallurgical examination of sections and surfaces.

Most of the nondestructive testing methods now available are discussed here, except visual inspection and radiography, which are so familiar to the nuclear industry that they need no elaboration. Many of the techniques are discussed in some detail in a publication of the American Society for Testing Materials.[22]

Dye penetrants are very useful in the detection of external weld defects, cracks, unbonded joints, pinholes, seams, or other discontinuities which are open to an external surface. The penetrant technique consists of completely covering the surface of the specimen with a highly penetrating liquid containing a visible or fluorescent dye. The liquid penetrates the open discontinuities; the excess penetrant is removed from the surface; and detection of a flaw is made by visual observation of the entrapped penetrant that bleeds out of the flaw. Some defects are so small that penetrants will not find them even though they may extend through the containing wall to the fuel. On the other hand, penetrants are useful in the detection of some discontinuities which cannot be found by any other technique.

Ultrasonic techniques may be used in a number of ways to inspect materials both before and after fabrication. In such inspections, high-frequency elastic vibrations are transmitted into the material, and an analysis is made of the vibration which is reflected from or transmitted through the material.[23] One use of this principle is the pulse-echo (reflection) technique for the detection of longitudinal defects in tubing.[24] Defects 0.0015 in. in depth and $\frac{1}{16}$ in. long can be detected on the inner or outer surfaces of small-diameter tubing. Pulse-echo techniques are also used to detect seams and cracks in the solid cores of uranium fuel elements prior to assembly. Ultrasonic velocity and attenuation measurements are made on these elements to assure dimensional stability of acceptable cores. Resonance ultrasonics can be used to measure the wall-thickness variations in tubing or cladding-thickness variations of unbonded fuel elements. A common use for ultrasonics is the detection of unbonded areas in bonded fuel elements. One technique uses the through-transmission principle, since the unbonded areas interrupt the passage of sound through the element. Decreases in the intensity of the transmitted sound are indicative of flaws. Some alternate techniques provide for monitoring the ultrasound which is reflected from the cladding; modifications or variations in the reflected sound may be indicative of unbonded conditions.

A poor bond between cladding and core may be detected with either thermal or electrical measurements. A common thermal technique involves coating the clad element with a temperature-sensitive material which will change its appearance as a function of heat absorption. As heat is applied to one side of the element, the unbonded areas will prevent heat transfer between the cladding and the core material and may therefore be delineated. In an electrical measuring technique that has been used for the detection of unbonded areas, a constant current is passed through the element between two current probes. A second pair of probes near the the current probes measures the electrical potential across the fuel element. During scanning, any perturbation in the potential is considered to be indicative of unbonded area or other defect.

Eddy-current methods are used in a number of ways to evaluate components and completed fuel assemblies.[25] The eddy currents are in-

duced in the part to be inspected by exciting a probe coil, which is applied to the part, with a high-frequency alternating current. The flow of eddy currents changes the impedance of the coil as a function of the conductivity, permeability, thickness, presence of flaws, or other variations in the specimen through which the eddy currents are flowing. Analysis of the impedance changes (amplitude or phase relations or both) can provide the desired information concerning the specimen. A similar technique may be employed with a separate pickup coil, or with an encircling coil rather than a probe coil. The eddy-current technique has been successfully applied to the inspection of tubing for the detection of cracks, intergranular attack, foreign-metal contamination, and, in some instances, dimensional variations. Effective use of eddy currents can be made in the measurement of cladding thickness of both bonded and unbonded elements.[26] This technique is also used for the detection of voids in the bonds of elements that contain a liquid-metal-bonding layer for heat transfer.

Another cladding-defect test consists of holding the fuel element under a high fluid pressure for some period of time, removing the assembly from the impregnation vessel, and quickly heating it to a temperature which will cause the fluid to have a high vapor pressure.[27] The existence of a leak may be deduced either by the presence of the released fluid or by the resultant distortion or damage to the defective assembly. An alternate technique consists of using gas as the pressurizing medium.[28] Sudden release of the pressure causes expansion of the flaw as the gas leaks from the pressurized element or causes swelling of the defective element. In order for this technique to be effective, the defect must extend through the containing material so that the fluid can penetrate into voids or porous material. This would be a nondestructive method on all but defective elements.

Helium leak-detection techniques are, in general, applicable to the detection of leaks in the cladding of unbonded fuel elements.[29,30] Mass-spectrometer helium leak detectors are available which can detect helium flows as low as 1×10^{-9} to 1×10^{-10} standard cm^3/sec. The general procedure is to fill the fuel-element voids with helium, either during fabrication or subsequently in a helium pressure chamber, and then measure the helium emission from the element. This can be accomplished by probe techniques (sniffing) or by placing the specimen in an evacuation chamber, the latter technique being much more sensitive. Further enhancement of the evacuation-chamber sensitivity can be obtained by allowing the leaking helium to accumulate in the evacuated chamber. Precautions should be taken to prevent removal of the helium during any evacuation prior to making the test.

Other nondestructive methods which are used for fuel-element evaluation consist of electrode potential testing[31] and the use of magnetic field meters.[32] Both can be used to detect the presence of foreign contamination which could cause failure in the element. The first method involves the electrode potential difference that exists between dissimilar ionic materials when immersed in a suitable electrolyte. The field meters are, of course, sensitive to ferromagnetic materials. Very tiny particles can be detected and precisely located by both techniques.

This is not a complete and exhaustive study of all nondestructive methods which are applied to fuel elements. Others have been successfully used to measure properties of interest, and, undoubtedly, new techniques will be developed to meet new requirements. The methods that have been discussed here are those which are now in general use. (R. W. McClung and H. J. Metz)

Reliability in Control and Electrical Systems

The fear of reactor runaways has been responsible for the development of highly reliable systems designed to release safety rods and thereby shut down the nuclear process upon the occurrence of an unsafe situation. These systems are characteristically continuously monitored, or tested, to ensure operability and are duplicated or triplicated, as may be required, to ensure adequate protection. It should be noted that a reactor core may be damaged quite as seriously by the result of inadequate cooling as by the overproduction of neutrons; however, in systems intended to protect against loss of coolant, the array of instruments, monitors, and testing routines is not usually so extensive as in systems for preventing the overproduction of neutrons. This may be accounted for in part by the availability of nuclear safety systems and in part by the greater fear of the more spectacular nuclear accident. In any event, the attention given to system reliability generally becomes

progressively less with increased distance from the reactor core.

A recent experience with an electrical power system for in-pile loops at the Oak Ridge Research Reactor (ORR) provides an example of decreased reliability where increased reliability was intended. The loops required a reliable electric power supply so that at least one coolant-circulating pump for each loop would continue to operate upon loss of utility power. It was feared that complete loss of coolant circulation when the experiment was in operation would result in meltdown of the in-pile sample, with attendant equipment damage and loss of data. A diesel-driven generator was available, but it would not have been satisfactory because of the time required for the machine to be brought up to speed upon loss of utility power. On the principle that greater reliability would be obtained from a machine which operated continuously, a battery and a dc–ac motor-generator set were installed. The a-c machine was connected to the experiment bus, and this bus was connected to the utility power system. During normal operation, the a-c machine was operated as a synchronous motor, with the battery floating on the terminals of the d-c machine. Upon failure of utility power, the d-c machine was to operate as a motor with power supplied by the battery, and the a-c machine was to operate as a generator to supply power to the in-pile experiments.

An inherent weakness of such a power supply lies in the interdependence of the motor-generator set and the utility system, which were both connected to the same bus. Upon the loss of utility power, it was imperative that the local breaker connecting the bus to the utility system be opened, otherwise the dc–ac set would attempt to carry the utility-system load remaining connected to the bus. Such a load could be equivalent to a short-circuit condition on the motor-generator set. The reverse-power relay intended to open the breaker was an important link in the chain, and failure of the relay could cause the failure of the entire power supply for the in-pile loops. In this example, failure did indeed occur.

The importance of the relay had not been recognized, as was demonstrated by the method used in testing the power supply, i.e., by opening the local breaker manually. No test was performed to establish that the local breaker would be opened automatically upon loss of utility power. When the utility power to the ORR building was interrupted, the reverse-power relay did not open the local breaker and power to the in-pile experiments was interrupted. Only after the failure had occurred was it discovered that a time delay of 5 sec has been incorporated in the response of the reverse-power relay. It was most fortunate that the interruption of utility power took place with the reactor shut down, and no damage to the loops occurred.

An increase in reliability could have been obtained in the electric power supply by employing two independent sources of power, with the utility power being one source and an alternator driven by an independent, continuously operating prime mover being the other source. Each source should supply power to separate coolant-circulating pumps. Maximum reliability is obtained when two power sources are cross-connected or made interdependent only through the process itself so that a failure common to both systems can occur only when the process fails.

In a proposal of means for assuring the safety and control of the 200-Mw(e) CANDU reactor, Siddall[33] has applied principles which might be adapted to a wide variety of systems wherein reliability is a requirement. In this proposal, control is effected by raising or lowering the moderator level with the use of the gas-balance principle. The calandria is connected to the dump tank by large dump ports having the geometry of inverted U tubes. The flow of the D_2O moderator into the dump tank from the calandria is controlled by the gas pressure maintained in the dump tank with respect to the pressure in the gas space above the moderator. Under normal conditions the moderator is pumped back to the calandria at a constant rate. However, since the balance of the gas pressure between the calandria and the dump tank controls the rate of flow from the calandria, the moderator level can be regulated to control the reactivity.

Failures of equipment would be infrequent in the proposed arrangement and would have little effect. Differential gas pressure would be maintained by three blowers connected in series so that the failure of one or even two of the blowers would not affect operation. Three pumps operating in parallel would return the moderator to the calandria to ensure operation as long as one pump continued to run.

A rapid dump of the moderator is obtained by means of six dump valves placed in three parallel gas lines which allow gas to flow from

the dump tank to the calandria. Relatively small valves can pass sufficient gas to obtain a satisfactory dumping rate. This arrangement permits any pair of valves comprising a trip channel to operate incorrectly without causing a shutdown, and it follows that the correct operation of the valves can be tested at will. Since the dump ports are simply inverted U tubes through which D_2O is continuously flowing, there is no problem of failure to open, failure to close, or of leakage to cause concern. Control is obtained with six control valves arranged in the same manner as the dump valves. Three pairs of valves are actuated by three independent control channels in such a way that any one of the three channels may fail without affecting control, which would be continued through the combined action of the remaining two channels.

In the main, Siddall's systems are successful because identical units are independent to the maximum degree, i.e., any one of the units may fail and yet the process will continue without perturbation and without the action of any device which is not already in operation. There are numerous examples in Siddall's proposed control system in which the principles of reliable systems have been applied, and this approach to system reliability cannot be dismissed as requiring the expensive duplication of components. Very often reliability can be achieved by the judicious use of inexpensive components, e.g., the small valves for regulating gas flow.

(E. P. Epler)

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Removal of Fission-product Activity from Gases

This is the second part of a two-part review* of the available information on methods for the removal of fission-product activity from gases. The previous issue of this Review[1] dealt with the removal of iodine, ruthenium, and particulate matter from gas streams; the removal of the noble gases and the design of multipurpose fission-product removal systems are discussed here.

Removal of Radioactive Noble Gases

The noble-gas fission products can be removed by charcoal adsorbers at room temperature, by refrigerated charcoal adsorbers, or by solution in liquids. The room-temperature process has the advantages, as pointed out below, of simplicity of operation and applicability to extremely radioactive gas mixtures, such as those generated by circulating-fuel reactors; but it has the disadvantages of large volumes and the need for fire-extinguishing facilities. The refrigerated adsorbers have the advantage of comparatively small volume and the disadvantage of complex refrigeration equipment. Provisions must be made to compensate for the loss of refrigeration, and the system must be designed to avoid the explosion hazard associated with the accumulation of oxidizing impurities.

The removal of noble gases by solution in liquid has the advantage of concentrating xenon and krypton to very high levels in a continuous process, but it has the disadvantages of large volumes, complex equipment, and, if intensely radioactive gases are present, the possibility of radiation damage to the solvents. Based on the information presented here, it appears that no practical process is available for the removal of xenon and krypton from a high-velocity gas stream, such as the total gas stream of a gas-cooled reactor.

Room-temperature Charcoal Adsorbers. The noble gases do not ordinarily participate in chemical reactions, and therefore it is necessary to employ processes such as physical adsorption or solution in liquids to separate them from other gases. When they are collected, the noble gases, unlike iodine or particulate materials, are not fixed in the collector. If the gas stream continues to flow, the noble gases will eventually be swept out of the collector. This may be prevented in either of two ways: in one case the retention time for the fission gas is made long compared with its half life; in the other case the collector containing the noble gas is removed from the process stream.

Adsorption on activated charcoal has been used as a concentration process in the production of xenon and krypton from the atmosphere, and a similar system is used for removal of krypton and xenon from the off-gas of the Homogeneous Reactor Test (HRT) at the Oak Ridge National Laboratory (ORNL). Based on equilibrium adsorption information, the HRT adsorber was designed[2] for operation at room temperature and to provide holdup times for krypton and xenon isotopes that would be long compared with their respective half lives, with the exception of Kr^{85}. Experiments conducted at ORNL to determine the dynamic adsorption behavior of krypton and xenon at very low concentrations under conditions similar to those in the HRT adsorbers indicated that the performance of the adsorbers would be adequate.[3,4] By November 1959 this off-gas system had operated over a period of two years during which the HRT had logged 350 Mwd of operation.

The fission-gas disposal system in the HRT consists of three adsorbers; two are normally used in parallel, and the third is reserved as a spare. Each adsorber contains 520 lb of activated charcoal in pipes starting at ½ in. in diameter and increasing in steps to 6 in. in

*In the first part of this article, which appeared in the March 1960 issue of this Review, a correction should be made on page 46, line 20, changing "filter more readily" to "filter less readily."

diameter at a point sufficiently removed from the reactor for the heat from radioactive decay to have decreased sufficiently to avoid excessive central charcoal temperatures. Oxygen and the fission-product gases flow through each adsorber at a rate of 1250 cm^3/min, with a normal inlet temperature of 25°C. The six-day retention time for krypton and the 60-day retention time for xenon are sufficient for virtually complete decay of all the fission gases, except Kr^{85}, which is vented to the atmosphere through a small stack in the exclusion area.

In other work at ORNL the retention times of experimental adsorbers were measured by a radioactive tracer technique, with Kr^{85} and Xe^{133} being used to typify the fission gases.[5] Pulses of the radioactive tracer gases were injected into the inlet gas stream, and the times of appearance of the tracer gases in the outlet gas stream were observed. The use of the pulse technique to simulate behavior of an adsorber in which the noble gas is entering continuously was justified by the results of a room-temperature experiment in which oxygen containing normal krypton at various partial pressures from 0 to 760 mm Hg was used as the sweep gas. The effect of krypton partial pressure was negligible up to 1000 μ Hg. The retention time was found to be proportional to the amount of charcoal and inversely proportional to the volume flow rate of the sweep gas at a constant operating pressure. The proportionality coefficient, called the dynamic adsorption coefficient, was evaluated for various adsorber materials and conditions. Activated charcoal was found to be the most effective adsorbent, with molecular sieve materials second best. Retention times were measured with helium, hydrogen, argon, air, CO_2, Freon-12, and krypton, individually, as the sweep gas. It was found that water vapor and CO_2 reduced the retention of krypton on the charcoal and that the retention time increased exponentially with the reciprocal of the system temperature.

For a given amount of activated charcoal and a given volume flow rate, it was found that long adsorbers are better than short adsorbers, even though average retention times are the same, because earlier breakthrough occurs in the short adsorbers. A theoretical expression describing the distribution of retention times was developed and experimentally verified, and the following expression that is useful for designing a practical adsorber system was derived:

$$m = \frac{Ft_b}{kt_b/t_{\max}}$$

where m = weight of charcoal
F = volume flow rate of the sweep gas
t_b = breakthrough time required for decay of radioactive gases or for cessation of gas flow
k = dynamic adsorption coefficient
$t_b/t_{\max}$ = ratio of the breakthrough time to the average retention time

The last two quantities are evaluated empirically. Typical values of k are 50 cm^3/g at room temperature for krypton in most gases, including air, and 1000 cm^3/g for xenon; $t_b/t_{\max}$ has the value of 0.6 for an adsorber 5 ft long and 0.8 for an adsorber 24 ft long.

The use of activated charcoal to remove radioactive xenon from the off-gas of an irradiated-fuel dissolving process has also been reported.[6] In an activated-charcoal column 5 in. in diameter and 25 in. long, containing Cliffchar charcoal with air flowing at an estimated 630 cm^3/min, breakthrough was observed at 120 min. The shape of the breakthrough curve was found to be in good agreement with that predicted by gas-chromatography theory. The breakthrough time was substantially less than that observed in similar tests at ORNL,[5] possibly because of differences in the charcoal used.

A proposed application illustrative of the use of a decay type of activated charcoal adsorber at room temperature was recently described[7] in which the radioactive volatile fission products are to be delayed for times that are long compared with their half lives in an activated charcoal bed which is to serve continuously without regeneration. The effluent gas from this system is to contain only Kr^{85} and is to be recycled as the inert gas for reactor operation.

Since activated charcoal is combustible in air or oxygen, the possibility of a fire must be considered when it is used. For example, an accidental fire occurred in a charcoal adsorber used in the off-gas system of an experimental assembly in the MTR.[8,9] In this experiment, air passed over an experimental fuel element in the reactor, through a heat exchanger and a particulate filter, and then vertically at 30 ft/min through two charcoal adsorbers in series, each 18½ in. in diameter and 5 ft long. At the time of the fire, the experimental fuel element had

been removed from the flux zone, and the external dose rate from the accumulated fission-product activity in the charcoal adsorbers had decayed from approximately 1000 r/hr to 200 mr/hr. The air flow through the in-pile facility had been continued, but the resistance to air flow in the system, including the adsorbers, had been gradually increasing and, just before the fire, had increased abruptly. As the air flow decreased, the temperature of the experiment piping in the reactor increased; however, at no time did the temperature of the air leaving the heat exchanger exceed 140°F. The charcoal ignited by an unidentified mechanism, and, as air continued to flow, a rubber gasket melted near the inlet at the bottom of the adsorber. A mixture of charcoal and air blew out through the crack where the gasket had been, burned vigorously, melted a lead shield surrounding the adsorber, and dispersed the radioactive contents of the adsorber throughout the MTR building. The fire was extinguished by halting the gas flow.

Following this experience, tests were conducted on the prevention and control of charcoal fires in order to evaluate the HRT off-gas adsorbers.[10] Ignition temperatures as low as 290°C were observed for charcoal in pure oxygen. With continued oxygen flow, temperatures in excess of 1000°C were observed in the center of burning charcoal beds; however, the fires did not damage the stainless-steel container tubes which were immersed in stationary cooling water. It was found that a fire could be extinguished by momentarily diverting the flow of oxidizing gas to an alternate adsorber. It was concluded that no serious consequences would result for the HRT design, provided that a fire was detected and extinguished within a few days. It was recommended that the charcoal adsorbers be equipped with temperature-sensing devices and that provision be made for diversion of flow of oxidizing gases in the event of a fire. In any such system the container should be designed to withstand the heat from a charcoal fire either by choice of materials or by cooling provisions, and there should be an excess of adsorber material to allow for the loss of charcoal by combustion.

During operation of the HRT in July 1958, a fire was accidentally started in one of the charcoal adsorbers by a failure of the catalytic recombiner which allowed an explosive mixture of hydrogen and oxygen to enter the charcoal adsorber. When the recombiner resumed operation, an explosion ignited the charcoal. The procedures developed experimentally[10] were applied successfully, and the released gaseous activity did not exceed permissible concentrations for xenon and krypton activity.[11]

Refrigerated Adsorbers. Since the adsorption of xenon and krypton increases as the temperature of an adsorber is reduced, fission-gas adsorption systems are often designed for operation at or near the temperature of liquid nitrogen. The principal benefit obtained is the small size of the adsorber, which is a factor of considerable importance in mobile installations. The design of a low-temperature adsorber is influenced by provisions for safety in the event of loss of refrigeration. Means must be available for stopping the flow of radioactive gas, and the systems must be designed to withstand the pressure buildup that will occur as the adsorber warms to room temperature. Precautions must be taken to avoid the accumulation in the adsorber of explosive mixtures of oxidizing gases, and water and condensable materials must be removed from the inlet gas stream to avoid plugging the refrigerated parts of the system.

The performance of the rare-gas processing plant at the Idaho Chemical Processing Plant (ICPP) is of interest because of the long experience with refrigerated adsorbers. As a result of a recent review, modifications were made for continuous rather than batchwise operation.[12] Oxidizing gases in the dissolver off-gas are reacted with hydrogen on palladium catalysts supported on alumina, and the resulting mixture of nitrogen, hydrogen, xenon, and krypton is passed through activated charcoal at the liquid-nitrogen temperature. The xenon and krypton are collected in bottles by raising the temperature of the adsorber. In operation, precautions have been required to control the composition of the gas in order to avoid the accumulation of explosive mixtures at various points in the process. Occasional freezeups have occurred when moisture accidentally entered the system, and liquid-nitrogen consumption has been rather high. On the whole, however, performance has been satisfactory. Experience with a small-scale apparatus for the purification and concentration of Kr^{85} in which charcoal is used as the adsorber at the liquid-nitrogen temperature has also been described.[13,14]

A recently designed fission-gas removal system uses silica gel as the adsorber at the tem-

perature of liquid nitrogen.[15] Silica gel was selected to avoid the possibility of a charcoal-oxygen explosion. The system was designed to handle helium containing 100 ppm krypton at a flow rate of 1 cfm. A 3-in. schedule-40 stainless-steel pipe filled to a length of approximately 46 in. with 3360 g of 4/16-mesh silica gel was specified for a krypton capacity of 665 g, which would correspond to approximately 1000 hr of operation at design flow. The system was designed for a pressure buildup of 1000 psi upon warming of the adsorber to room temperature. Limited tests were conducted on a prototype model. The use of silica gel appears to avoid many of the hazards associated with the potential explodability of charcoal if oxidizing impurities accumulate. Caution will have to be used in silica gel systems, however, to avoid the accumulation of radiation-produced ozone in the adsorber.

The N.S. *Savannah* is provided with a refrigerated charcoal adsorber to retain gaseous fission products if there should be leakage from any of the clad fuel elements in the reactor.[16] The system was designed and built by Arthur D. Little, Inc., for the Babcock & Wilcox Company. It is designed to process, at the rate of 0.72 cfm, hydrogen with fission-gas contamination at 120°F and 45 psia. Each of three adsorbers is to provide a 50-day holdup time for krypton when maintained at a temperature of 100°K. Oxygen is removed catalytically to prevent the accumulation of explosive concentrations of oxygen in the charcoal adsorbers. The breakthrough of Kr^{85} will be monitored periodically by analyzing the exit gas. The system is designed to withstand a pressure buildup of 1200 psi at room temperature. In a recent review of several ozone explosions in cryogenic apparatus in radiation fields,[17] it was concluded that explosions could be avoided if oxygen and/or ozone were excluded from the low-temperature regions where, in static systems, they would tend to concentrate by liquefaction. On this basis it was recommended that the initial oxygen content of the liquid-nitrogen coolant be less than 200 ppm.

An explosion occurred at ORNL in August 1958 in a refrigerated charcoal adsorber that was being used to concentrate radioactive krypton and xenon for use as radioisotopes. The rare-gas concentrate, as received from the ICPP, contained air, xenon, krypton, and NO_2, and it was drawn through a charcoal adsorber at −80°C. When the adsorber was allowed to warm to −10°C, an explosion occurred that split the 2-in.-diameter adsorber vessel near its entrance and blued the iron parts near the split, indicating a high temperature. The system had been designed to withstand a pressure of 8000 psi. It was theorized that an exothermic reaction yielding nitrogen and CO_2 had occurred between adsorbed NO_2 and carbon. The system had operated safely during prior runs but at lower concentrations of NO_2. The system was redesigned, and provisions were made for passing the gas mixture over carbon at 950°C before entering the adsorber. No further explosions occurred after this modification was made.[18,19]

Solution in Liquid. A process for the recovery of fission-product noble gases by solution in liquids has been developed at the Brookhaven National Laboratory (BNL).[20] The solubilities of gaseous xenon and krypton at low partial pressures in a carrier stream, such as nitrogen, were determined in various solvents by a dynamic-tracer method. The gas mixture containing radioactive tracer was bubbled through 26 different solvents until equilibrium was reached, as evidenced by equal inlet and outlet activity count rates. The amount of tracer retained in the solvent was calculated from the integral of the concentration of the effluent gas. The results indicated that the solubility of xenon and krypton increased as the solvent became less polar. The solubility data go through a maximum in the vicinity of a cohesive energy density value of 8.0, as expected from Hildebrand's theory of the solubility of nonelectrolytes.[21] Kerosene-based liquids, such as Amsco 123-15 and Ultrasene, were found to be the best solvents for this application. In Amsco the solubility of xenon decreases 4.5 times as the temperature is increased from 24 to 150°C, and that of krypton decreases 2.95 times over the same range. Henry's law is applicable. The solubilities, in cubic centimeters of gas per cubic centimeter of liquid, for xenon and krypton, respectively, were: in water, 0.10 and 0.05 at 30°C; in Amsco, 3.39 at 21°C and 0.56 at 25°C; and in Ultrasene, 3.53 at 21°C and 0.58 at 23°C.

A countercurrent absorption column 1.5 in. in diameter packed with ¼-in. ceramic Raschig rings was also tested using nitrogen sweep gas and liquid Amsco as the solvent.[20] Transfer-unit heights (a measure of column efficiency in which greater efficiency is indicated by smaller

heights) of 0.33 to 0.51 ft were found for gas streams containing 432 to 959 ppm by volume of xenon in nitrogen and up to 0.85 ft for krypton content in the range 41.6 to 6300 ppm. A design was described for handling, at the rate of 5 cfm, gas containing 300 ppm of krypton and xenon from a plant processing 1 metric ton of fuel per day. Absorber and stripper towers 30 in. in diameter by 24 ft high packed with 1-in. Raschig rings would be provided to carry an Amsco flow of 66.5 gal/min at room temperature. Ninety-nine per cent of the krypton and 100 per cent of the xenon would be recovered with a concentration factor of 4.0.

In a later report of work at BNL,[22] improvements in the continuous, selective, solvent-absorption process for three different solvent-gas systems are described. The three systems are kerosene-CO_2, liquid and gaseous N_2O, and Freon with noble gases. By addition of a third column, i.e., a fractionator, between the absorber and stripper and an increase of the design pressure to 10 to 15 atm, it was possible to reduce the volumes of circulating liquid and to increase the product concentrations, and better solvents were found. Solubilities are given in the report[22] for krypton, xenon, nitrogen, argon, and oxygen in liquid nitrous oxide and in Freon-12 at approximately $-85°C$ and higher temperatures.

The kerosene-CO_2 system operates at room temperature and uses a caustic solution to remove the CO_2 from the product at the end of the process, leaving a product containing 77 per cent noble gases. Such a system to process the off-gas from a 1 metric ton/day fission-product plant, at a flow rate of 10 cfm, requires three columns 16 in. in diameter by 18 ft high. The process has the advantage of operation at ambient temperature, but the CO_2 cost is a consideration.

The liquid nitrous oxide process operates at temperatures down to $-98°F$ and is primarily of interest where the feed gas contains nitrous oxide as a major impurity. It uses three columns 10 to 12 in. in diameter by 18 ft high and allows elimination of a nitrous oxide removal operation, but the hazard of using a thermodynamically unstable solvent must be considered.

The Freon process operates at temperatures down to $-94°F$ and gives a product containing 85 per cent noble gas. Three columns 10 in. in diameter by 18 ft high are used. The xenon recoveries are 99.9 per cent, and the krypton recoveries are approximately 98.4 per cent. This process offers the advantages of low refrigeration requirements, low solvent cost, and maximum stability.

A similar process, developed in England, uses carbon tetrachloride as the solvent.[23] Air, together with nitric acid fumes and fission gases from a fuel-element dissolver, is fed at a rate of 6.5 liters/hr to a column $1\frac{1}{2}$ in. in diameter by 6 ft high packed with $\frac{1}{4}$-in. porcelain rings. The column operates at room temperature with a continuous liquid carbon tetrachloride phase which flows at 36.5 liters/hr. The liquid phase is then stripped in another column with carbon tetrachloride vapor, which is obtained by boiling 5 per cent of the liquid flow to give a feed rate of 6.5 liters/hr. An enrichment factor of 70 is obtained for krypton, and 99 per cent of the krypton is recovered. Much of the product gas is N_2O, which can be removed chemically for obtaining a higher enrichment factor. The theoretical transfer-unit height is 18 in. It is concluded in the report[23] that radioactive krypton from the dissolution process can be concentrated into a small volume in one absorption column of moderate size.

Other Techniques

The possibility of separating gaseous fission products from gases by thermal diffusion was investigated in the Netherlands.[24] Experiments were run on the separation of Xe^{133} from deuterium by thermal diffusion, and in one experiment a practical thermal-diffusion separation system was simulated on a reduced scale. A column 75 cm long by 0.26 cm in diameter was used with a platinum wire 0.04 cm in diameter extending along its axis. The column was filled with a mixture of deuterium and Xe^{133} at pressures up to 1 atm, and the wire was heated to provide an average gas temperature of 450°K, with the wall temperature at 300°K. Thermal-diffusion parameters were evaluated by observing the initial rate of change of xenon concentration at the end of the column. The experiment was conducted without gas flow. Separation factors from 200 to 1800 were observed with relaxation times for thermal-diffusion equilibrium (i.e., time to reach $1/e$ times the equilibrium concentration) of 400 to 800 sec. The observed relaxation time was about 0.6 to 0.8 of the calculated relaxation time, indicating that the theoretical equations are somewhat pessimistic. Based on the results of this test, a system was

studied for the separation of Xe^{133} from deuterium at a pressure of 40 atm. Two concentric cylinders would be used, with the inner cylinder having a diameter of 2.56 cm, the annular spacing being 0.29 cm, and the over-all length being 272 cm. The inner cylinder would be heated to a temperature of 250°C, and the outer cylinder would be kept at 30°C. It was calculated that the column would be able to process 15.8 liters (NTP) per hour and would decrease the xenon concentration by a factor of 10 in the product. The xenon-rich concentrate would have a volume three to five times smaller than the feed-gas volume. The performance of such a separation system was judged not to be good enough to justify its complexity.

When radioactive impurities appear in a gas stream flowing at a low rate, for a limited time it is possible to store the gas in large containers for decay and subsequent release to the atmosphere during favorable meteorological conditions. This procedure has been employed at the ICPP by using a 10,000-cu ft storage tank for the gas evolved during caustic dissolution of fuel elements.[25]

A similar procedure is proposed for use with the Enrico Fermi fast-breeder reactor, where four storage tanks are provided to contain 250 cu ft of gas at 515 psia. The gas is to be cooled in the tanks for 56 days before discharge.[26] A similar procedure is used at Shippingport.[27]

Multipurpose Fission-product Removal Systems

Designs of several off-gas systems for use with reactors or with in-pile loops have been described in recent review articles,[28–30] and the off-gas system for the Plutonium Recycle Test Reactor (PRTR) was described in a recent Hanford report.[31] The PRTR is a 70-Mw vertical-pressure-tube D_2O-moderated reactor enclosed in a containment vessel which is ventilated without filtering through a 150-ft stack. The two areas of greatest air-borne contamination potential, a fueling vehicle and a fuel examination facility, are equipped with separate ventilation exhaust filters which discharge into the main exhaust system. These filters are Cambridge absolute filters Model 1B-600-0, with a design collection efficiency of 99.95 per cent for a particle size of 0.3 μ or larger, and Model 1A-1000, with an efficiency of 99.97 per cent for the same conditions. Air-sampling taps are located downstream of each filter for intermittent air sampling. There are also provisions for continuous sampling of the stack exhaust at the 50-ft level, and a stack exhaust filter can be installed at a later date if needed. The containment vessel, including the exhaust duct, would be completely sealed in the event of high activity at the base of the stack.

A system for holdup of fission gases was designed for an ORNL experiment in the MTR.[32] A refrigerated charcoal adsorber was provided to process helium flowing at 200 cm^3/hr and carrying all the gaseous fission products from a 10.4-kw fused-salt-fueled in-pile loop. Another system was designed to process, at a rate of 5600 cm^3/hr, nitrogen which would normally contain no fission gases. The helium system required 2.4 kg of charcoal and the nitrogen system, 4.3 kg. Calculations were made of the amounts of xenon and krypton activities which would emerge for various modes of operation at −30°C and at 25°C. It was concluded that the helium system could operate continuously without breakthrough of radioactive gases and that the nitrogen system could operate for 5 hr at −30°C. Both these gas systems operated satisfactorily throughout two in-pile experiments.

Another fission-gas holdup system was evaluated for use with capsules operating at 4.4 kw of fission power in the ORR.[33] The gases to be processed were the mixtures of nitrogen and helium, at a pressure of 285 psi, that were used in checking clad capsules for leaks. A charcoal adsorber 4 in. in diameter by 15 in. long, containing 1000 g of activated charcoal, was specified to provide a 30-min krypton holdup time, a 5-hr xenon holdup time, and an iodine-removal efficiency of 99.9 per cent. The gas flow rate was 900 cm^3/min. The amounts of krypton, xenon, and iodine isotopes released by various modes of operation were calculated.

A fission-gas holdup system was also considered for an ORR fused-salt capsule in-pile experiment,[34] and calculations were made of the consequences of accidental release of fission products from a capsule containing 26 mg of U^{235}. Radiation doses to the personnel on the ground were calculated for krypton, xenon, iodine, and their radioactive daughters, and it was concluded that disposal of the krypton and xenon through a 250-ft stack would not exceed permissible tolerances. It was recommended that an iodine trap having an efficiency of at least 90 per cent be used.

A fission-gas processing system was designed for a 10-kw homogeneous reactor type of in-pile loop operating in the ORR.[35] A unit adsorber trap containing 440 g of 14/18-mesh Columbia G activated charcoal was specified to provide a holdup volume for krypton of 6.6 liters of air sweep gas. Various series and parallel arrangements were made of unit traps to meet capacity and flow requirements of various parts of the off-gas system. A bed of Linde molecular sieve material was incorporated for moisture adsorption. The amounts of various activities that would be released by various modes of operation were calculated.

A helium-purification system recently proposed for the EGCR is of interest, although it is not principally concerned with the removal of radioactive material from gases.[36] The recommended purification processes include a catalytic converter for the oxidation of CO, H_2, and hydrocarbons to CO_2 and H_2O by O_2 gas; cooler condensers for the removal of the bulk of the H_2O; silica gel adsorbers to complete the removal of H_2O; and Linde molecular sieve adsorbers for the removal of CO_2. The gas-purification system was designed to process 180 lb of helium per hour at 300 psia and 859°C. The system was designed to limit the steady-state impurities to 18,000 ppm CO_2 by weight and 1000 ppm H_2 by weight for a system having a total helium flow of 4.27×10^5 lb/hr.

The design of an addition to the central off-gas system facility at ORNL was recently described.[37] The system addition will process the radioactive gases from chemical processes and from the ventilation of process vessels in various locations. The present 2000-cfm capacity is being increased to 6000 cfm. The gases to be processed are corrosive because of their acid content, they are radioactive, and they contain only small quantities of particulate matter. All principal items of equipment, except the blowers, are enclosed by shielded walls with means for performing routine operation without entering the enclosure. Steam-powered blowers and solution pumps are provided with automatic changeover equipment that will be activated by failure of electric power. The gas to be processed is passed through a caustic scrubber packed with Saran or Dynel fiber and washed with 5 per cent NaOH. There are three stages of absorption and two stages of demisting. A steam-heating coil is provided to prevent condensation downstream. The gas then passes through 4-in.-thick silver-coated copper mesh, fire-resistant absolute filters, and an additional 4-in. layer of silver-copper mesh. The gas stream then enters the 250-ft-high stack and is added to the 120,000-cfm air flow in the stack. (W. E. Browning)

Containment by Conventional Building Structures

Most existing or planned power reactors employ a steel containment shell to prevent the escape of large amounts of radioactivity in case of accidents. There is a great incentive to modify or eliminate this pressure shell to avoid the additional expense or at least to reduce it. A novel scheme used at the ORR employs a conventional building of steel-sandwiched insulation panels to contain fission products in case of core meltdown.[38] In this facility no attempt has been made to seal the building completely. Instead, automatic equipment is provided to maintain a flow of air into the building in the event of an accident and to exhaust the building air–fission product mixture through filters and scrubbers to remove particulate activity and the halides. The cleaned gases would be exhausted up a 250-ft stack. In this way the fission products would not be contained as in a pressure vessel; rather, the activity would be processed and released from the stack at a controlled rate.

The various criteria to be applied when evaluating this type of containment for a particular reactor system can be illustrated by considering the case of the ORR.[38] This reactor is designed to operate at a power level of 20 to 30 Mw. The maximum credible accident was taken to be the release of all the volatile fission products contained in the core after it had been in operation for 39 days at 20 Mw. It was assumed that the gases would mix uniformly with the building air and that the building would remain intact. The fission products in the gas cloud that would be biologically important would include Kr^{85}, Kr^{87}, Kr^{88}, Rb^{88}, Kr^{89}, Xe^{133m}, Xe^{133}, Xe^{135}, Xe^{135m}, I^{131}, I^{132}, I^{133}, I^{134}, and I^{135}. It was assumed at the time the analysis was made that the scrubber could remove 99.9 per cent of the iodine and all the particulate matter. (Subsequent tests have shown that this equipment will actually remove 99.0 per cent of the iodine.)

The principal biological internal hazards are associated with iodine and with Sr^{89}, which de-

cays from Kr^{89} through Rb^{89}. In the analysis, the maximum permissible concentration[39] of Sr^{89} for continuous occupational exposure was taken to be 2×10^{-8} $\mu c/cm^3$, whereas total dose resulting from the inhalation of 25 μc of iodine (equivalent to approximately 25 rem to the thyroid[40]) was considered to be acceptable. The maximum ground concentration obtained from the 250-ft stack emitting 1 curie/min with a wind velocity of 1 mph may be calculated from atmospheric diffusion formulas[41] and is 3×10^{-7} μc per cubic centimeter of air. With the use of the above values, the maximum possible doses as a function of exhaust rate were determined. It was found[38] that an exhaust rate of less than 12,000 cfm would yield an iodine dose that would be less than the maximum acceptable dose.

The Sr^{89} activity posed a more difficult problem. Following the same procedure as for iodine, a curve was developed to show the exhaust rate (cubic feet per minute) versus the time (hours) during which the maximum permissible Sr^{89} ground concentration would be exceeded. It was found that, at an exhaust rate of less than 1800 cfm, the maximum permissible Sr^{89} ground concentration would never be exceeded. On the other hand, it was found that an exhaust rate of about 5000 cfm would give the longest time interval (2.8 hr) during which the maximum permissible concentration would be exceeded. Higher exhaust rates would decrease this time interval, but there would always be a finite time period during which the maximum allowable concentration would be exceeded. However, for accidents such as that considered here, activity concentrations in excess of the maximum permissible are expected, and the analysis would have had more significance if the results had been expressed in terms of the total inhalation exposure, as was done for iodine.

The external biological hazard from the exhaust cloud was also calculated, and it was found that the maximum permissible dose of 25 rep would not be exceeded at flow rates of less than 20,000 cfm. As would be expected, the calculations showed that the direct radiation from the building would be formidable, i.e., initially about 47 r/min at a distance of 100 ft from the building. This hazard, however, is essentially independent of the type of containment and can be accommodated only by shielding or by providing a controlled exclusion area around the site.

Reactors larger than the ORR with higher power levels and longer fuel-irradiation times would present more stringent conditions. However, the analysis of the containment system would proceed along the same lines. The allowable rate of activity release would, of course, have to be based on human tolerance to the more biologically critical radionuclides. The building would have to be sufficiently tight to ensure a negative gauge pressure across the walls at the release rate specified. Methods similar to those developed by Luckow and Patterson[42] can be used to analyze and predict building leakage.

In a system of this type, particulate filters can be used to remove practically all nonvolatile fission products. Iodine can be removed to almost any desired concentration[43] by proper choice of chemisorption materials, but krypton cannot be chemically held up and would have to be discharged to the atmosphere, after which it would decay to strontium. This may be the fundamental limiting factor in this kind of containment. If, however, the gases could be held up to allow sufficient time for decay of the krypton to ruthenium and strontium, the activity could be effectively filtered. Since the decay chain is

$$Kr^{89} \xrightarrow[2.6\ min]{\beta^-} Rb^{89} \xrightarrow[15\ min]{\beta^-} Sr^{89} \xrightarrow[53\ days]{\beta^-} Y^{89}$$

it is evident that the gases must be held up some multiple of 2.6 min, depending on the maximum permissible concentration of Sr^{89} in the atmosphere.

Holdup could be accomplished by constructing a double building around the reactor in which the inner walls would contain the reactor gases. The annular area between the inner and the outer walls would be swept by inleakage from the outside, and the gas flow would be filtered and sent up the stack. This method would also serve to dilute the effluent gases. Such dilution might, however, decrease the effectiveness of the iodine-removal system.[43] The noble gases could also be held up in charcoal-adsorption pits (see page 47 of this issue), if the flow required to maintain a negative pressure in the building could be maintained at a reasonable pressure drop.

If the reactor contained high-pressure gas or water, a loss-of-coolant accident could cause a rapid increase in pressure in the containing building. This pressure would have to be vented to prevent destruction of the building. For reactors with clad fuel elements, a time lag would

exist between the loss of coolant and the escape of fission products due to melting of the cladding. In some reactors, this time interval would be sufficiently great to allow the direct release of the initial surge of relatively uncontaminated gas to the atmosphere through some sort of louver arrangement that could afterward be shut sufficiently tight for the blower exhaust system to maintain an inflow of air. Another scheme[44] considered is the use of an inertia plug to absorb the initial pressure loading; subsequently the gas would be directed to a holdup volume and then exhausted through filters and up the stack.

Filters can be utilized to remove particles of almost any size by proper staging of various-sized filter pads. One of the more promising systems is a CWS filter followed by a series of wetted Saran or Dynel fiber packs, the fiber size in each stage being determined by the range of particle size to be removed in that stage.[42] Iodine can be removed with very good efficiency by filters of silver-plated copper mesh. Such filters are ideally suited to applications where large volumes of gas must be handled with a low pressure drop. A practically unlimited iodine-removal efficiency can be obtained by backing these filters with charcoal adsorbers, if the penalty in pressure drop can be accommodated.

A comparison of the cost of this type of containment with that of the more conventional steel shell is difficult because some items essential to one type but not the other are frequently required in the facility design for other reasons. In particular, the building containment system requires both an exhaust system equipped with a good filter system and a stack to effect the high-level release of the effluent activity. These items may or may not otherwise be required by that facility. On the other hand, were the same nuclear facility to be provided steel-shell containment, a comparable filter system might not be required, nor would the stack be required in order to limit the accidental release of activity. However, a filter system within a containment vessel is desirable, if not always necessary, and a stack is also useful when subsequently venting the contained activity and is probably required for the normal discharge of activity from the facility.

The cost of the ORR building was somewhat less than is usually estimated for a structure of this type, largely because of the lack of windows. The cost of a comparable containment vessel for the ORR has been estimated, on the basis of equal volume, to be $650,000 (this is for an 800,000-cu ft spherical vessel which is designed for an internal pressure of 16 psig and which includes air locks). On the other hand, a comparable cylindrical vessel large enough to contain the basic ORR facility with adequate headroom would consist of a cylindrical vessel 75 ft in diameter, 85 ft in cylindrical height, with hemispherical heads, and would cost $500,000. In this instance $90,000 was added as a separate item for 6000 sq ft of offices and shops in a separate building adjacent to the vessel.

A comparison of the cost of containing the ORR with both containment techniques is shown in Table IV-1. This table indicates the possible range of some of the principal costs associated with both types of containment. Although the cost of pressure-shell containment ranges considerably higher than that for the building containment, the two ranges do overlap so that it is not possible to conclude that the building containment will be cheaper in all situations where both techniques are applicable. Rather, each situation will have to be evaluated in terms of (1) the maximum credible accident associated with the reactor in question and (2) the facilities, i.e., stack and ventilation equipment, which are otherwise available at the installation and which might be exploited with one or the other of these containment techniques.

In Table IV-1 the building containment costs are real, having been taken from actual ORR construction figures. These figures did not permit the breakdown of cost further than that indicated here. Neither of the containment costs includes that of a crane, although the building containment costs do include structural support for the crane, whereas the pressure-vessel costs do not. On the other hand, the costs of locks and some penetrations are included in the pressure-vessel costs but not in the building costs. The special vent louvers and external doors for the building containment are included under the item "isolation provisions and control equipment." This item under pressure-shell containment includes only isolation valves for lines and ducts penetrating the vessels; the maximum and minimum estimates of these penetrations are 120 and 60. Two stack costs are included in the estimates; one is for a 250-ft-high, 9-ft-ID brick stack similar to those which are frequently employed in nuclear facilities, and the other is for a 250-ft-high, 3-ft-ID metal

Table IV-1 COMPARATIVE COSTS OF BUILDING AND PRESSURE-SHELL CONTAINMENT FOR THE ORR*

Item	Building containment			Pressure-shell containment	
	Maximum	Actual	Minimum	Maximum	Minimum
Container, i.e., building structure or containment vessel	——→	$350,000†	←——	$ 650,000‡	$500,000‡
Auxiliary building					90,000
Scrubbing system	——→	75,000†	←——	30,000	
Isolation provisions and control equipment	——→	120,000†	←——	170,000§	85,000§
Stack	$150,000¶		$ 56,000**	150,000¶	
Totals	$695,000		$601,000	$1,000,000	$675,000

*These values do not include escalation.
†Reference 45.
‡Reference 46.
§Personal communication from C. Cooper to M. H. Fontana, May 4, 1960.
¶250-ft-high, 9-ft-ID stack; personal communication from G. Morris to M. H. Fontana, May 4, 1960.
**250-ft-high, 3-ft-ID metal stack; personal communication from C. E. Winters to W. B. Cottrell, May 4, 1960.

stack (3 ft is the minimum diameter needed to pass the necessary ventilation air from the building containment). A simple recycle filter system is included in the maximum costs for the containment-vessel technique, although this is frequently not required.

Thus the controlled-leakage type of containment is not a panacea; it must be evaluated for the given conditions. First consideration must be given to the determination of the amount of fission products released during an accident. Also, power reactors must be carefully analyzed to evaluate the effect of pressure surges on the enveloping building. Also, a determination of the sensitivity of the fuel cladding to the maximum credible accident would be required. A reasonably slow pressure rise could be accommodated by means of a vent system, but a blast wave would require attenuation. The allowable gas discharge rates determined from biological considerations would have to be balanced against the required tightness of the building to infiltration of air under the worst possible wind conditions. Costs to be considered would include filters, air-cleaning equipment, blowers, and automatic control equipment. If the plant were to be located in or near large centers of population, the containment building would have to be designed so that maximum permissible accidental exposures would not be exceeded. The feasibility of the economical use of this type of containment for a given reactor depends on the particular conditions. In any case, a scheme of this type should be considered along with the conventional steel-shell concept.

(M. H. Fontana)

Reactor Materials as Chemical Poisons

Many materials used in and produced by nuclear reactors are, as is well known, hazardous because of their radioactivity; a considerable number of such materials are also chemical poisons. Safety of personnel from the chemical effects of this broad class of materials is generally assured if proper protection from the associated radioactivity is provided. Accordingly, special handling of materials *after removal from a reactor system* to prevent *chemical* poisoning of personnel is not generally a problem in itself.

On the other hand, special handling of materials in research, development, and testing programs designed to provide reactor materials is often required. It is well known that many common nonradioactive materials are toxic if ingested or inhaled in substantial quantities. A survey of such toxicity and of the generally simple rules and precautions for safe handling of these many materials is clearly beyond the scope of this publication. It is obvious, however, that reactor materials which may require more than ordinary care in handling to assure safety of personnel are of concern to nuclear

safety. Some recent studies of the toxicity of beryllium and its compounds and of the possible hazards from exposure to polyphenyl moderator-coolants are discussed here.

Beryllium and Its Compounds

The toxic nature of beryllium and several of its compounds has received general recognition within the past 15 years. It is known that both the handling of beryllium and its compounds in powder form and the breathing of air contaminated with beryllium or its compounds represent distinct hazards.

Dermatitis can result from contact of crystals of a "soluble" beryllium compound with an open wound, from exposure to fumes or dust of soluble beryllium compounds, or from insoluble beryllium compounds embedded under the skin. Either an acute or a chronic disease, designated berylliosis, may result from inhalation of beryllium or its compounds. The acute disease has been caused by inhaling as little as 45 μg of beryllium (as BeF_2) over a short interval.[47] The chronic disease can result from retaining material over long periods. In this case the symptoms may appear several years after the last exposure.

The beryllium materials which have thus far been found to be dangerous are Be, BeO, $Be(OH)_2$, $BeSO_4$, BeF_2, $BeCl_2$, and $MnZnBeSrO_4$.[48] No cases of the acute or chronic disease have been reported for exposures to beryl ore or to alloys containing less than 2 per cent beryllium. An exhaustive review of the natural history and medical studies of beryllium has recently been published.[49] This document summarizes the clinical findings from case histories, traces environmental research, and includes a bibliography of important papers on the subject.

Warren and Ferris have presented, in a brief report,[50] a summary of safety precautions, operating techniques, and monitoring methods to be used in a special ORNL laboratory facility for studies of the reprocessing of beryllium-containing reactor fuels. Control of beryllium handling at the British atomic energy installation at Harwell has recently been described by Brooks.[51] On the subject of Maximum Permissible Level Recommendations, this document states:

> In December 1948, a panel comprising members from the Department of Research in Industrial Medicine, the Medical Research Council, Ministry of Labour and National Service, the Atomic Energy Research Establishment and Ministry of Supply met in London and recommended for A.E.R.E. Harwell that "the atmospheric pollution of workrooms should not exceed a provisional level of 1 μgm. Be/m^3 of air."
>
> Since October, 1949, an amended level not exceeding 2 μgms. Be/m^3 for the average concentration during the 8-hr. working day has been in use at A.E.R.E. on the advice of the above panel. This level had already been adopted by the United States Atomic Energy Commission on the recommendation of their Medical Advisory Committee on Beryllium.
>
> To cover transient exposures within, and pollution of air near beryllium plants, this committee later recommended in addition that:
>
> (i) "Under no circumstances and at no time should the concentration of beryllium in the air exceed 25 μgms./m^3, even though the daily average (i.e. 2 μgms. Be/m^3) might be satisfactory." (The U.S.A.E.C. Memorandum of Recommendations for the Control of Beryllium Hazards, issued to all contractors by the Division of Biology and Medicine in August, 1951, differs from this by stating that "no personnel should be exposed to a concentration greater than 25 μgms./m^3 for any period of time, however short").
>
> (ii) "In the neighbourhood of an A.E.C. plant handling beryllium compounds, the average weekly* concentration at the breathing zone level should not exceed 0.01 μgm./m^3."
>
> and these were also adopted by the U.S.A.E.C.
>
> These two latter recommendations were not officially adopted at Harwell, but were noted, and since that time [it] has been the practice at the Atomic Energy Research Establishment to ensure that these are also kept.
>
> *raised to "monthly" in March 1950.

Brooks[51] has included brief discussions of types of beryllium materials handled and their end uses, procedures for transporting, storing, and ultimate disposal of beryllium compounds, and procedures for decontamination of apparatus and equipment. Also, the general considerations that have been applied at AERE installations in the design and construction of beryllium-handling facilities are described. Either brick or plaster walls covered with a high gloss paint, waxed linoleum floor coverings faired at edges and corners for easy cleaning, and bench tops of smooth slate or of stainless steel are recommended. Special attention has been paid to design of exhaust hoods and glove boxes and to ventilation systems for these laboratories. Air

samplers from which an average concentration of beryllium over the sampling interval can be obtained are described, as are direct-reading beryllium monitors from which a continuous record of air contamination is available. In general, it would appear that the precautions used in handling beryllium and its compounds in Great Britain are very similar to best practices used in the United States.

Organic Moderator and Coolant Materials

The considerable success of polyphenyls as moderator-coolants for nuclear reactors, such as OMRE, has prompted investigations of the possible hazards resulting from exposure to these and related materials. The results of such a study have recently been published.[52]

The compounds investigated were monoisopropylbiphenyl, with and without reactor irradiation; a terphenyl, with and without reactor irradiation; and unirradiated samples of biphenyl, *o*-terphenyl, *m*-terphenyl, and *p*-terphenyl. The method of Draize et al.[53] was used to study ocular and skin irritation in rabbits and intradermal irritation and sensitization in guinea pigs. The method of Spiegl et al.[54] was used to study the pulmonary effects produced by 1-hr exposure to each of the polyphenyls at two different concentration levels in air.

Only monoisopropylbiphenyl and the terphenyl mixture were irritating to the conjunctiva in rabbits, and only the unirradiated moderator-coolants caused skin irritation in rabbits. The authors state: "Both the moderators and their components were highly damaging to guinea pig skin following intracutaneous injection. All of the polyphenyl compounds except irradiated monoisopropylbiphenyl produced sensitization and chemical necrosis. The latter produced only necrosis. Monoisopropylbiphenyl and ortho- and meta-terphenyl were the only polyphenyls that caused death after inhalation, although all of the compounds produced some of the following symptoms: nasal congestion with rhinitis, lachrymation, labored respiration, and erythema of the ears and paws. The following histopathological changes were also observed: acute tracheal necrosis, acute tracheobronchitis, pulmonary edema, broncheopneumonia, atelectasis, and petechial hemorrhages." The authors conclude[52] that "all such toxic effects can be prevented by protective clothing and respirators."

(W. R. Grimes)

Radiation Monitoring of Liquid Wastes

An atomic plant, like most industrial plants, uses large quantities of water. Pollution of the water by radioactive materials constitutes a special hazard to plant personnel and to the general public when the water is returned to the public domain as process waste. Extensive studies have consequently been made of means for detecting and controlling this special hazard. Instrumentation presently available ranges from simple monitors that energize alarms at some set point of gross activity in a contaminated stream on a "safe-unsafe" basis to complex instrumentation that makes a rather complete analysis of the activity and initiates control action.

In order to orient this discussion of radiation-monitoring procedures and problems, assume a waste-handling system designed to process the waste water from a facility that includes reactors and a spent-fuel processing plant. Waste water is defined as nonrecirculated reactor cooling water and process water from the spent-fuel processing plant. The waste system therefore handles high-flow, relatively low-activity water under normal operating conditions. Abnormal conditions might raise the activity level by a factor of 100, and the system must be capable of handling such slugs of activity. Typically, 5×10^5 gal of water with activity levels ranging from 10^{-4} to 10^{-5} μc/ml might be handled per day. A special piping system collects the water from the process areas and conveys it to some central point, such as a settling basin, from where it may be released to the public domain through direct release, seepage from pits, or a treatment plant. The goal of the system is to provide sufficient holdup and treatment capacity to ensure that the radioactivity in the effluent to the public domain is no greater than the recommended maximum permissible concentration (MPC).[55] Since the recommended values cover specific radionuclides, the MPC levels range over five decades; and since the averaging may be done for a period of as long as a year, the logical method of monitoring at the downstream end of the system, before control is lost, is by taking samples and performing laboratory analyses to determine concentrations of at least the most dangerous radionuclides. The laboratory analyses customarily consist of radio-

chemical analysis and energy spectrometry. Although the state of the art is sufficiently advanced to permit designing and building automatic equipment to do the sampling, analyzing, and compiling of the data,[56] the problems of cost, maintenance, and calibration are formidable. Manual sampling for laboratory analysis is therefore the usual procedure.

The total flow of the contaminated water is also of interest, since the samples must be representative and proportional to the total liquid, particularly if they are to give an accurate composite over an averaging period (typically, 30 days). In addition, data on the total water flow and its variations are usually required for the control of water consumption and for process control. The downstream end of the system must therefore be provided with flow measurement and proportional sampling instrumentation. Weirs or flumes, in conjunction with level recorders, are used to measure total flow, and automatic samplers of several types are available.[57] Measurements of water level behind a weir can be made by using a float-actuated chart instrument. Proportional sampling is commonly done with a continuously or intermittently revolving scoop which is shaped so that the scooped sample is proportional to the total flow through the weir. When a pressure differential is used for level measurement, the chart instrument may have an integrator added so that a pump takes the sample by running at a speed proportional to total flow.

The contaminants generally found in waste effluents are fission products, although other contaminants, such as the heavy elements from uranium up, are not uncommon. The fission products are usually beta- and gamma-activity emitters, and the heavy elements are alpha-activity emitters. The monitors commonly used are Geiger-Mueller tubes with thin shells of about 30 mg/cm^2 of material. These monitors are useful for beta particles of energies of 0.5 Mev or higher, such as those from most fission products, but the detection of lower energy activity, such as that of Pm^{147} or alpha particles, calls for a different detection method. The first point in the waste stream where total flow is available upstream of treatment and holdup facilities is an appropriate location for alarm monitoring for alpha and soft-beta particles, as well as for beta-gamma monitoring, in order to obtain an indication of abnormal amounts of activity in the process waste as early as possible.

Monitors for Gamma and Hard-beta Activity

In waste-handling systems such as the one in operation at ORNL[58] (see Fig. 1, page 84 of this Review, for a flow sheet of this system), the waste water passes through an automatic diversion valve after being sampled and monitored. The monitor[59] that operates the valve bases its control action on the activity level of the controlling nuclide,[55] which, in this case, is Sr^{90}. The effluent is normally released to a settling basin and thence to an open stream, but, if the level of activity exceeds 150 counts/min/ml, the effluent is automatically diverted to the waste-treatment plant. The effluent from the waste-treatment plant is periodically sampled before release. If the activity is less than the MPC, the effluent is discharged to an open stream; if not, the effluent may be recirculated through the waste-treatment plant.

All continuous water monitors are plagued with background buildup as a result of deposition on the walls of the liquid container, and therefore the ORNL monitor was designed to cope with this problem.[59] The use of plastic liners made from inexpensive plastic bottles provides a smooth surface that decreases deposition and eases the problems of decontamination. The liners are easily replaced and may be discarded. The water stream itself serves as shielding from activity buildup at other points in the system. The sampling flow rate is 1 gal/min, and the monitor has a minimum sensitivity of 6.8×10^{-6} μc/ml. The allowable MPC is 10^{-7} μc/ml, but, as mentioned above, there is additional treatment, i.e., at least settling and dilution, before control is relinquished. Experimentally the sensitivity limit corresponds to 15 dis/min/ml for Sr^{90}-Y^{90} in equilibrium. Modifications for increasing the sensitivity of this monitor have been studied.[59] Interesting features of the monitor are the design of the lead shield which permits the top to be moved aside for maintenance and the use of a dead zone in the control circuit to prevent the valve it controls from cycling when the radiation level is fluctuating about the set point. A more recently described British instrument uses many of the same features.[60]

Monitors for Alpha and Soft-beta Activity

Alpha and soft-beta activity, which is not detected by beta-gamma monitors and which might be present in curie amounts, is normally de-

tected by periodic or proportional sampling followed by laboratory analysis, although automatic sampling and analysis for gross alpha and beta activity have been effected at Hanford[61] and at Savannah River.[62] The poor penetrating properties of alpha and soft-beta radiation makes it desirable to have the material in as close contact as possible with the counter, and evaporation of the liquid sample allows close coupling as well as the removal of the shielding by the water. A machine was devised to automate the process; this machine forms "dimples" in a strip of aluminum tape, pipets a precise sample (about 2 ml) into the depression, evaporates the sample to dryness, and passes the sample to a thin-window beta counter and then to an alpha counter. The Hanford machine operated some 1500 hr, with a sensitivity of at least 2.58×10^{-5} μc/ml. The Savannah River model is still in operation, although maintenance is difficult and it may soon be replaced with a "swirl" cell of the type described below.[63] The dimpler used at Savannah River takes five samples per minute as compared with one in 10 min for the Hanford model. At Savannah River the sample size is 1.5 ml, and the detectors are both scintillation counters. The sensitivity is not defined in the report, but alarm levels are specified as five times background for beta and gamma and 20 times background for alpha. These levels correspond to 7.3×10^{-3}, 3.3×10^{-3}, and 5.2×10^{-5} μc/ml, respectively. Although it is admittedly complex mechanically, this approach does permit almost continuous monitoring of soft-beta and alpha activity, since alarm times are of the order of 3 min.

Another approach to bringing the counter and the fluid into close contact has been described by Hurst.[64] The fluid is made to swirl on the walls of a container so that a thin sheet of water surrounds a counter without touching it. Although the problem of background buildup is still present, the water wall can be made thick enough to shield the counter. Hurst has also used the swirl technique to count alpha particles in a special thin-walled flow counter, the construction of which is described in reference 64. This counter, which is useful in a laboratory, is unfortunately not satisfactory in the field. The use of a thin-walled geiger counter in the swirl cell, however, makes a rugged field instrument without moving parts. A variation of this technique is reported by Wingfield.[63] In order to emphasize his improved design, Wingfield defines a low-level detection limit as the minimum concentration to yield, within 1 min, a signal-to-background ratio equal to or greater than 1 and a percentage deviation equal to or less than 0.05. A comparison of Hurst's swirl cell, the Savannah River dimpler, and Wingfield's swirl cell shows the respective sensitivities to be 7.3×10^{-4}, 1.4×10^{-3}, and 3.5×10^{-5} μc/ml for Sr^{90}.

Other methods that have apparently not been extensively adopted for avoiding background buildup and still reducing the distance between the water surface and the counter are (1) to use a tray of moving water and (2) to position the counter in the center of a falling cylindrical curtain of water formed by many water streams.[65] Mention should also be made of a "dip" monitoring scheme, which consists of immersing in the water a commercially available detector that is protected with a disposable plastic bag. An ultimate sensitivity of 4×10^{-7} μc/cm^3 for $Sr^{90} + Y^{90}$ is quoted for the detector.

Tributary and In-line Process Monitors

Monitors placed even closer to the source of potential activity in the waste-water system are of lesser complexity and importance, and are used primarily for process control. Although these monitors are not specifically essential to safety in the handling of waste water, they merit mention. Beta-gamma monitors on tributary streams enable plant operators to locate the exact source of an increase in activity levels in waste streams without delay. In-line gamma and alpha monitors are of interest because of background buildup at high levels of activity. Pillinger[66] has described an alpha monitor for such service. Plutonium solutions with activities up to 10^8 dis/min/ml were used to test the properties of a Teflon liner for the monitor. The counter always read within ±10 per cent of the correct alpha level regardless of what activity had been accommodated in prior usage, and, by pretreatment with a solution containing the maximum activity anticipated, the background was stabilized at about 1 per cent of the total count produced by the pretreatment solution.

Wingfield[67] has described a small cell with a Teflon liner, which, in use, is flushed with nitric acid to reduce the background buildup. A measure of the seriousness of the buildup problem is the fact that the background increased by a factor of 4 in five days of operation. An accuracy of 10 per cent is quoted for comparison with

samples checked by laboratory analysis. Mackey[68] has used two other systems. In one, the sample holder is a fluorothene tube with a highly polished bore (15-μin. finish); in the other, the stream is allowed to free-fall past the detector and thus bypass the walls to preclude buildup. Operation of the fluorothene-tube model has been successful, but the falling-stream model has not been fully exploited in the reviewer's opinion. The model that was built did not provide enough shielding between the headpot, where the falling stream was formed, and the detector, and thus the background buildup was excessive.

Fuel-element failures have been detected in very short times by monitoring the reactor coolant for fission products, but, although this application is highly instrumented, it is not discussed here because at least one comprehensive review of the subject is already available.[69] Two bibliographies with sections on monitoring instrumentation are also available.[70,71]

(D. J. Knowles and F. E. Gillespie)

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Geologic and Hydrologic Considerations in Power Reactor Site Selection

It is required that a reactor-hazards report provide, among other things, the meteorological, hydrological, geological, and seismological data necessary for evaluating the measures proposed for protecting the public against possible radioactive hazards. Since meteorologists had their previous experience with industrial air pollution to guide their early work with respect to the hazards of radioactivity, the new methods of analysis that they developed during their studies of reactor sites were immediately applicable to reactor-hazards analyses.[1] Studies of the portions of current hazards reports[2–20] that deal with hydrology, geology, and seismology show, however, that earth scientists must learn to provide a more complete picture of the physical environment if they are to discharge their responsibilities. The various hazards reports do not approach the problems in the same way, and the differences are only in part due to differences in the sites.

Despite the differences, the discussions are of two general types. In the first, the ways in which the site might affect the reactor are analyzed; and, in the second, the ways in which the reactor might affect the surrounding area are considered. Few reports attempt to cover both aspects of the problem.

Effect of Site on Reactor

The reports that are primarily concerned with the effect of the site on the reactor include studies of foundation conditions, safe bearing loads, and results of the special test drilling done to obtain information about these points. Several reports are almost exclusively concerned with such data. They were prepared, in general, by engineers in the regular employ of the petitioning company. Such investigations are, of course, routine for any large, heavy structure, but a power reactor is more than that and raises special problems.

Another old but less routine problem is presented by the possibility of floods. Most power reactors are situated near some large body of water for condenser cooling, and in several cases the reactor is within a very few feet of past or predicted maximum flood or high-tide levels. The economic desirability of a low pumping head is obvious, but the art of flood prediction has repeatedly fallen victim to what one distinguished hydrologist called "the concatenation of circumstances." Flood crests far higher than the predicted maximum have not infrequently confounded the best of prophets. None of the reactor sites appears to be in danger, but in at least one case the margin of safety appears to be less than that used to guard against some of the more spectacular hazards. In any event, the flooding of a reactor does not seem to offer any serious hazard to the surrounding area unless other misfortunes occur at the same time, as is, of course, possible.

The possibility of earthquake damage has aroused considerable interest,[21] probably more than it generally deserves. Most of the hazards reports provide the information required on this subject by a short statement that the site is in an area where earthquakes are infrequent and of low intensity, and therefore the hazard from them is negligible. Such statements are usually based on data taken from publications of the U. S. Coast and Geodetic Survey. In any case, reactors are constructed so that they are not susceptible to damage from any but the most intense shocks. There might, however, be an indirect hazard through damage to underground pipes or wires outside the reactor or through landslides or tidal waves, either of which might cause flooding; but, even including these somewhat remote possibilities, the total earthquake hazard is very small in most areas.

An exception to this generalization would be the case of a reactor built by the water's edge on weak foundation material in an area of frequent earthquakes of high intensity. One suggested

site nearly meets these conditions. It is on the edge of a bay and is underlain to a depth of 20 ft largely by recently deposited unconsolidated beds of clay and silt. The nature of the material below this is not clear. In the geology section of the safeguards report,[11] the material is described as "slightly consolidated gravels, sands, and clays with the majority consisting of the finer grained materials"; that is, largely clay and silt. In the section on earthquake hazards, however, the same material is referred to as rock and is described as a "fairly well indurated series of mudstones, siltstones, sandstones, and conglomerates." The drilling record calls the material clay and sandy soils with some sand and gravel.

There is also some ambiguity as to the intensities of the earthquakes to be expected. The section on earthquake hazards lists eight shocks of intensity VIII (modified Mercalli scale) and one of intensity IX over roughly the last hundred years, but the shock of intensity IX is discounted because the record is old and uncertain. The authors conclude that intensity VIII is the maximum shock which should be expected in the future. The ambiguity is in the word "expected." With at least nine shocks of intensity VIII a matter of record, more shocks should certainly be expected, but the report appears to give the impression that there is no need to plan for anything of greater intensity. There is a very great difference between a reactor situated on rock and subjected to an earthquake of intensity VIII and one situated on water-saturated clay and silt and subjected to a shock of intensity IX. The principal hazard in this case appears to be not to the reactor but to the proposed underground waste-storage facilities.

Shock intensities VIII and IX are defined as follows:

> VIII. Damage slight in specially constructed buildings; considerable in ordinary substantial buildings, with partial collapse .... Sand and mud ejected in small amounts. IX. Damage considerable in specially designed structures, great in substantial buildings, with partial collapse .... Ground cracked conspicuously. Underground pipes broken.

In general, it may be stated that the site studies which have tried to answer the question "is the environment safe for the reactor?" have been able to furnish convincing reliable answers, very largely because the problems are not new. The foundation problems even for a large reactor are minor compared with those encountered in building a large bridge or dam and can be solved by almost routine methods. Flood prediction is a little more difficult, but the reactor as such offers no novel problem in raising the question: "How much can we cut the pumping lift and still be safe above flood crest?" Earthquake damage is far more dramatic, and because no area can be said to be quite free of the danger of a heavy earthquake, there has been much public interest. The danger to a well-built reactor situated on firm foundation material appears, however, to be negligible.

Effect of Reactor on Site

On the other hand, the reports which interpret the question of site evaluation as "what will a reactor accident do to the surrounding area?" deal with a very complex and novel problem, as has been suggested by the several attempts to establish criteria for site evaluation.[22,23] The hazards reports in which an attempt has been made to evaluate in any way the vulnerability of the surrounding area to release of radioactive materials give data relevant to some parts of the problem, but in no case has an attempt been made to follow a nuclear incident through to its conclusion and evaluate its consequences. Admittedly, such an analysis would require the joint efforts of a group of experts, but the largely uncoordinated and separately composed sections which comprise the typical hazards report of today do not seem to provide as complete a basis as could be wished for such a study. Even with the best available data, however, the attempt would be faced at every step by problems for which there are at present not even order-of-magnitude solutions.

The analysis should probably begin with the maximum credible accident, already a time-honored maneuver, even in this infant industry. Next, having decided how much of what particular isotopes would escape and in what physical and chemical form, the meteorologists would have to decide where the air-borne material would land and the hydrologist would have to plot the paths, rate of movement, and dilution of any escaping liquid. Some of this liquid might seep into the ground, and rain might leach radioactive material from the fallout area and carry this also into the ground water, so the direction and rate of movement of the ground water would have to be considered. Next, the ecologist would have to attempt to determine

how the radioactive materials would move out of the lithosphere and hydrosphere into the animal and plant life of the area; and, finally, the health physicist would have to attempt to assess the damage. Like so many "councils of perfection," this abbreviated outline of the required task is at present seldom, if ever, followed, and yet it is essential to the preparation of a complete hazards report.

It is unfortunate that the apparent emphasis in hazards studies on the problem of ground-water contamination has tended to draw attention away from the problem of contamination of surface water. Geologists and hydrologists are, of course, familiar with the problem of sea-water encroachment into underground aquifers, on which there is an extensive literature. A well-known example of this condition exists in Brooklyn, N. Y., where the ground water is so contaminated with salt that the water-supply wells had to be abandoned, and in Atlantic City, N. J., where a considerable expenditure has been necessary to obtain sufficient fresh water. The hazard to water resources posed by a nuclear reactor is, however, quite different and is governed by other factors.

Almost all the fission products are cationic and are strongly adsorbed by the clay minerals, as has been shown not only by laboratory studies but also by years of waste-disposal operations at Hanford and Oak Ridge.[24,25] At these installations many millions of gallons of medium- to low-level waste, much of it high in inert dissolved solids, have been put into the ground; and yet hazardous concentrations of the radioactive constituents are not found beyond a distance of a few hundred feet. This is a far more severe test than any ground-water aquifer could ever be subjected to as the result of a reactor accident; and, except under rather special circumstances, the danger of any contamination moving any appreciable distance underground via the ground water does not appear to exist. The emphasis placed on this hazard appears therefore to be unjustified. Very few geologists have worked for any length of time at a site where substantial quantities of activity were being put into the ground and have, therefore, missed the field experience which would help to emphasize the differences between salt water and radioactive materials as contaminants.

An exception to the statement that ground-water contamination is not a problem must be made in the case of terrain underlain by cavernous limestone or certain types of lava flows. Water can move through these rocks rapidly and with a minimum of contact between liquid and rock. The site of the National Reactor Testing Station in Idaho is underlain by basalt flows through which the ground water moves almost as through a pipe; but even here the problems raised are principally in connection with waste disposal and not with reactor operation. Waste-disposal operations in general, because of the volume and nature of the liquids involved and the large inventories of long-lived fission products which may be present, could well pose potential ground-water contamination problems where reactor operations would not.

It is of interest to consider two of the fission products, iodine and ruthenium, which are not strongly adsorbed by the soil and which, because of their volatility, might escape in the event of a reactor accident. Elemental iodine is not particularly soluble in water and apparently alters to a soluble form in the soil rather slowly as compared with the eight-day half life of I^{131}. The movement of soil moisture from the surface to the water table and the lateral movement of the ground water below the water table are very slow in most areas, in general less than a foot a day, and only under very unusual circumstances could iodine travel through the ground and then become a hazard. Much the same can be said for ruthenium. Ruthenium-106 has a half life of about one year, but it is adsorbed to some extent by soil and therefore moves less rapidly than the ground water. Experience at Hanford and Oak Ridge shows that the Ru^{106} concentration after traveling even a short distance underground is greatly reduced. Even at that, however, Ru^{106} moves farther and faster than most of the other radionuclides and may well represent in many cases the principal potential hazard to ground-water supplies. This danger appears to be very much less, however, than the danger to surface-water supplies from radionuclides moving over the land surface.

The danger to water resources from a power-reactor accident is, in the opinion of the reviewer, almost entirely a surface-water problem. If the activity is released as a liquid, the important hazard will be from that part which can flow overland or through drains directly to a river, lake, or bay. The part which filters into the ground can almost be neglected unless its route is short, direct, and through particularly large openings. If the released radioactive

material is air-borne, the first concern must be with that fraction which falls directly into some body of surface water, and the presence of a large body of surface water in the prevailing downwind direction would clearly be important. Fallout on land also poses many problems, but, in general, it appears unlikely that the subsequent contamination of ground-water resources will be the most serious.

The important consideration for the hydrologist is not how the water moves but how particles will be carried by the water, since radioactive materials leached from the fallout activity will, in general, be quickly adsorbed by the soil. A favorable circumstance would exist if all the fallout occurred on a sandy soil into which all the precipitation would rapidly filter with no direct surface flow to the streams. The opportunity for ion exchange is, in general, better and lateral movement of ground water is less in a thick, permeable, sandy terrain than it is in an area underlain by clay or fractured rock. An unfavorable circumstance would exist if the fallout were onto a clay soil, particularly if the land surface were steeply sloping, since there might be rapid washing away of the soil.

The movement of radioactive materials in ground and surface water has been observed at Oak Ridge. Process waste water containing trace amounts of cesium and strontium was, for some years, discharged into a small stream. Relatively little of the radioactive material reached the Clinch River, into which the stream flowed a few miles away, since most of it was trapped on the mud in the bottom of a small lake through which the stream drained. The radioactive materials largely moved with the mud, not with the water. The lake has since been drained, and the adsorbed strontium is very slowly leaching from the sediment. The leached activity does not, however, travel far without being caught again.

The movement of radioactivity was also observed at ORNL in the fall of 1959, when maintenance work on exhaust fans caused the release of small particles of iron rust up a stack. The particles were spread over several acres downwind (see page 85 of this Review for a discussion of this incident). The rust contained ruthenium, which shortly showed up in water samples from streams draining the area. The soils in the Oak Ridge area are largely clay; the rocks beneath are, in general, of low permeability; and most of the rain flows directly off over the land surface and into the streams. At Hanford or Brookhaven, in contrast, the terrain is very permeable, and rain soaks into the ground where it falls. There is no direct surface runoff to the streams which, locally, are fed only by ground water, and the downstream contamination observed at Oak Ridge would not have taken place. It must therefore be concluded that geologically and hydrologically Hanford and Brookhaven are more desirable reactor sites than Oak Ridge. There are, however, many other factors which have to be considered before making a final evaluation for reactor siting.

Radioactivity in the mud on the bottom of a river or lake is not in itself a direct hazard, but the radionuclides are available to plant and animal life in the river and will slowly find their way into a series of complex interlocking food chains. This is, of course, the province of the ecologist, but the geologist can contribute to the problem by predicting the movement of the mud downstream, onto the river banks in time of flood, and eventually into some distant lake or bay or beach deposit. But the final evaluation must be based on ecological studies such as those of Kornberg relative to the Columbia River,[26] in which the fate of neutron-induced radioactive materials introduced into the river by reactor cooling water is traced.

One point raised by the advocates of an impermeable terrain is the relative ease of monitoring surface-water supplies as compared with the difficulties of monitoring ground water. The interpretation of ground-water samples is indeed a complex problem and should probably be termed "monitoring" only in the case of work with pumped wells fully penetrating a homogeneous aquifer. However, those very qualities of an impermeable terrain which keep the water on the surface where it may be easily inspected also keep water-borne contamination on the surface where it is most likely to get into one of our food chains. Granted, it is an advantage to be able to follow the movement of the radioactive materials. The ease of monitoring does not appear, however, to be an adequate measure of the relative hazards of the two types of terrain. If surface-water contamination is indeed a more important factor than ground-water contamination, a permeable terrain would be preferable in order to promote infiltration of precipitation and to minimize direct mud-laden surface runoff to the streams. These same factors are of considerable interest in con-

nection with flood control and soil-erosion control. Studies have shown these factors to be subject to considerable artificial control.[27] This suggests that terraces and small check dams, along with careful preservation of the plant cover adjacent to and downwind from the reactor site, could materially reduce the hazards in areas where the present conditions are unfavorable. To the extent that this approach is feasible, it would change the viewpoint from "what is the site now?" to "what might be made of it?"

Conclusions

In summary, the impressions gained from a study of the sections on geology and hydrology of most of the power-reactor-hazards reports prepared to date are as follows:

1. The problems have not been approached in a consistent manner.
2. Foundation, flood, and earthquake hazards, which are old engineering problems, are adequately covered, except as the degree of hazard may require large margins of safety.
3. The discussions of environmental contamination have overemphasized the danger to ground-water resources, and the stress on the desirability of an impermeable terrain has neglected the added degree of danger which this may create with respect to surface-water supplies.
4. There has not been sufficient recognition of the fact that, once out of the reactor, most of the radioactive materials will be attached to solid particles and will move with them rather than in solution.
5. The hazards posed by the reactor to the surrounding area can be evaluated only by a study of many interlocking factors, including the reactor itself, and to set up criteria for individual factors is a poor approach.

Progress will be made only as the various disciplines are coordinated to determine what the results of an accident would be and how much harm the results would cause. The studies by the U. S. Geological Survey of diffusion and sediment transport in rivers and the ecological studies at Hanford and Oak Ridge are important steps in this direction, but there is still much to be considered in evaluating the hazard that a power reactor would truly represent at any site.

(W. de Laguna)

Management and Disposal of Radioactive Wastes

The radioactive wastes being produced in atomic energy operations are classified according to level of activity, i.e., low, intermediate, and high, and the techniques for their management and disposal vary accordingly. Low-level wastes are generally considered to be those which can be discharged to the environment with no treatment or with only moderate treatment to reduce the radioactivity. Intermediate-level wastes are those with radioactivity content such that efficient control and intensive treatment are required before release. High-level wastes are those resulting primarily from first-cycle separations in the chemical processing of irradiated reactor fuels, and they may contain up to several thousand curies of radioactivity per gallon. Minor amounts of high-level solid and gaseous wastes must also be handled.

As set forth in Congressional hearings[28,29] on Industrial Radioactive Waste Disposal, the philosophy followed throughout the entire industry is to contain the high-level wastes on site in underground concrete and steel tanks and to disperse low- and intermediate-level wastes to the environment under stringent control. Each site maintains an extensive monitoring program to determine that the control of dispersal is adequate, since the radioactivity concentration must be below the maximum permissible concentration at the point of usage of the medium to which the radioactivity is dispersed. The storage, treatment, and monitoring procedures used at the various sites are discussed here.

High-level Wastes

High-level wastes exist as liquids that must be stored in tanks and, occasionally, as highly radioactive solids that must be handled by special methods, such as storage in vaults or land burial encased in heavy concrete, and as high-level process off-gases that require special air-cleaning facilities in connection with the process operation. Some 65 million gallons of high-level liquid wastes are presently stored in underground tanks of steel and concrete, primarily at Hanford, the Savannah River Plant, and the National Reactor Testing Station in Idaho. The practice of tank storage of liquid wastes is considered a temporary expedient until safer and

more permanent means of disposal are developed. Even if tank storage should prove to be the most economical means for the management of high-level wastes, the life expectancy of the present tanks may be only a fraction of the hazard-life of some of the fission products they contain. Therefore several methods for ultimate disposal are being studied in various nuclear energy research and development programs.

In addition to the potential radioactivity hazards, storage of high-level wastes in tanks involves other problems. One of these is the production of heat by radioactive decay which, in a concentrated waste, can cause the liquid to boil for long periods, perhaps a hundred years. However, for newly generated liquid wastes, tank storage for a period of years has the advantage of providing time for the decay which will reduce the radioactivity and ease the problems of heating and external radiation in ultimate disposal operations.[30]

Possible methods that have been considered for the permanent disposal of high-level wastes include storage in cavities in natural salt formations; injection through specially constructed wells into deep permeable formations in which natural conditions would provide storage space and prevent migration of the wastes; deep-well injection into impermeable formations after they have been fractured by hydrostatic pressure to form storage spaces for a slurry containing the waste liquids; concentration and fixation of waste constituents in a stable, solid medium, such as a ceramic or glass, and storage or burial in selected locations; and calcination, along with subsequent satisfactory storage.[31] Research and development studies on the methods mentioned above are being actively pursued by the AEC and other agencies. Assuming that the specific technical problems can be overcome, it appears that disposal in salt formations, fixation in solid media, and calcination offer most promise of early development; deep-well disposal holds more long-range possibilities when larger volumes of liquid wastes have accumulated and require ultimate disposal.[31–33]

Preliminary discussions and calculations have indicated the possibilities of deep-ocean disposal. Ocean disposal for high-level wastes is not considered to be a likely possibility by authorities in the United States but appears to be more attractive to a number of foreign investigators, particularly in countries with limited land area and a high population density. Inherent disadvantages of ocean disposal are insufficient quantitative oceanographic information to permit definition of the hazards, technical problems in carrying out ocean disposal operations, and the loss of control of the radioactivity once it is dispersed in the sea. Because disposal to the ocean is of interest in coastal areas of the United States and in a number of foreign countries and because this method is and may continue to be used for the disposal of packaged intermediate- and low-level wastes, physical and biological oceanographic studies are being made in order to ascertain the status of present ocean disposal operations and the feasibility and limitations of ocean disposal from the health and safety viewpoints. Much time will be required to accumulate this type of oceanographic information, but its future importance in relation to radioactive waste disposal is recognized.[31,34,35]

Salvage and utilization of valuable waste by-products have been considered as a possible solution to waste-disposal problems in many industries; and this has been proposed in the case of high-level liquid radioactive wastes. By-product recovery and sale with possible profit or at least partial recovery of the cost of operation is an attractive idea, but this method only alleviates to some degree and does not solve the problem of waste disposal. Selective removal of strontium and cesium from high-level waste streams represents a potentially important contribution to ultimate disposal schemes. If these two isotopes could be removed, the problem of heat generation in the wastes would be substantially reduced. The feasibility of disposal of the remaining material would be increased, and the economic return from utilization of the fission products would provide an incentive to prospective chemical-processing and waste-disposal operators. However, two points must be emphasized with regard to the possible extraction and utilization of specific nuclides. First, for waste-disposal purposes, removal must be extremely efficient; that is, decontamination factors of 10^6 to 10^8 are needed; and, second, the removed nuclides must eventually be disposed of when they are no longer usable (references 31, 33, 36, and 37). In an editorial review in *Power Engineering*,[30] these considerations are summarized as follows:

> Utilizing Radioactive Wastes. Some people believe that the problem of high-level radioactive

wastes may be solved by salvaging fission-product isotopes for beneficial use in industry and agriculture, in biological studies, and in medical services. To a certain extent this is possible. Many isotopes could be salvaged, but not all of the long-lived ones are useable now, and none are economically recoverable. For example, cesium and the rare earths may be separated and recovered, but this would not eliminate the need for storage of the residual waste for at least years or decades. Their removal would lower the radioactivity by a factor of 10,000 to 1,000,000, but the activity in the residual liquid would still be 1000 to 100,000 times greater than could be released to the environment.

Low- and Intermediate-level Wastes

Low- and intermediate-level wastes exist as liquids, as solid radioactive residues and contaminated solids, as air-borne radioactive gases, and as particulate matter. The liquid wastes result from second- and third-cycle separations of fuel-processing operations and from dejacketing. Low-level liquids also are derived from reactor cooling systems, laboratory operations, low-level processing, laundry operations, decontamination of equipment, and various discharges of slightly contaminated waste water.[31]

For low-level wastes, the disposal program has consisted of dispersal to the environment (water, ground, or air), with treatment if necessary. Low-level releases in various amounts are the principal source of environmental contamination from nuclear energy operations. This problem may be expected to increase as the nuclear power industry increases in size or if acceptable limits of radioactivity in the environment are further reduced.[29] At small sites, some liquid wastes are concentrated in volume, packaged, and shipped for disposal by land burial or burial at sea.[31] In larger operations, the greater volumes of liquid and air-borne wastes and the bulk of the solid low-level wastes make "absolute" treatment and containment prohibitively expensive.[29] Consequently the degree of treatment is determined by the amount of radioactive material that can be safely discharged to the local environment.

Because of inherent differences in the sites and kinds of wastes, the disposal practices followed at the several operating areas vary considerably. Standards for the release of wastes specify that the average levels of contamination in water and in air be lower than the accepted maximum permissible concentrations (MPC) in these environmental media. Examples of direct discharges to surface waterways are the release of large volumes of reactor cooling water containing some 60 measurable induced radionuclides into the Columbia River at Hanford[38] and the release of low-level wastes containing fission products to the Clinch River at Oak Ridge after treatment to remove most of the strontium, rare-earth, and cesium activity.[39,40]

At nearly all sites, some low-level wastes are released to the environment by percolating them into the earth at or near the surface. Such wastes work their way slowly into the ground water, leaving all or part of the contained radioactivity on the soil. Considerable use has also been made of this method for the disposal of intermediate-level wastes at Hanford and at Oak Ridge.[24,25] Authorities who have considered ground disposal by seepage near the surface have concluded that for the disposal of considerable quantities of radioactive materials this method involves unacceptable risks for general use because of uncertainties in defining underground conditions and in predicting the movement of radionuclides when diluted in the ground water.[41]

Established burial grounds for solid radioactive wastes exist only at the large production and research sites, such as Oak Ridge, Savannah River, Idaho, Los Alamos, and Hanford. Solid wastes from other sites are usually packaged and shipped for burial in these established burial grounds.[31] Some disposal of packaged radioactive wastes at sea has been made by sites near the Atlantic and Pacific Coasts, perhaps to the total extent of 15,000 to 20,000 curies. Usually the packages are heavy drums, containing wastes and concrete, which are dumped into the ocean at selected locations where depths are 1000 fathoms or greater.[31,34] For the control of air-borne gaseous and particulate radioactive materials, the practice is to discharge the materials through tall stacks for dilution in the atmosphere. Air-cleaning procedures consist of the use of high-efficiency filters, deep-bed sand and fiber filters, and iodine and rare-gas removal units.[31,42]

The acceptability and assurance of the long-term safety of release of low-level wastes to the environment, even when guided by accepted standards and continued monitoring determinations, are very dependent upon accurate knowledge of the environmental factors and how they

will affect conditions at the particular location. For this reason extensive research and evaluation studies have been conducted at specific sites and are being continued by geologists, hydrologists, meteorologists, oceanographers, ecologists, and soil scientists. The knowledge available in these special fields is essential as a basis for estimating the dilution or concentration of radionuclides that may be expected and for defining the behavior and ultimate fate of hazardous waste materials in this environment after dispersal.[43]

Environmental Monitoring

The controlled release of low-level wastes from various nuclear energy operations, the possible leaching or relocation of small fractions of high-level wastes from underground storage sites, and the accidental, irregular release of activity from nuclear energy operations will continue to pose problems from the standpoint of environmental health and safety. Those responsible for waste disposal at nuclear energy sites have a responsibility to keep waste releases within the limits of the accepted standards and to assure that no sources of potential human exposure exceed permissible concentrations. In addition to monitoring for protection of employees, it is necessary to maintain data from the environs for distances as great as the potential hazards might spread from the types of release discussed above. Primarily for the protection of the public, but also as proof that its responsibility has not been neglected, each site must maintain an extensive monitoring program to determine that the control of radioactive waste dispersal is adequate.[29]

The amounts of effort and funds devoted to offsite monitoring in connection with nuclear energy operations vary widely, depending upon the size of the operation, the nature of the hazard, the probability of dispersion, the population that may be exposed, and many other factors. At large installations the monitoring programs are very extensive and involve tremendous amounts of detailed work. For example, at the Congressional hearings in 1959, a representative of the Hanford operations submitted the following list of "environs measurements (annual)": total samples, 40,000; total analyses, 75,000; field measurements, 40,000; gaseous waste samples, 10,000; liquid waste samples, 17,000; vegetation and ground—samples, 11,000, and field measurements, 17,000; biological monitoring—birds, 500, rabbits, 700, and aquatic organisms, 1200.[38] Since 1943, numerous test wells for monitoring underground soil contamination have been installed at Hanford, with a total of 120,000 ft of well borings.

From the results of the Hanford monitoring program, it is estimated that the actual radiation exposure to persons in the environs of the Hanford plant is between 3 and 15 per cent of the currently accepted limits for the general population and that, for exceptional cases in which unusual amounts of local fish and leafy vegetables are eaten, the exposure may reach as much as 50 to 60 per cent of the appropriate limits. It was pointed out that these values include not only the contributions of radioactive materials from the reactor effluent and the separations-plant stack gases but also from world-wide fallout. It was stated that the deposition of isotopes on local vegetation from fallout exceeds that which originates from the Hanford plant.

The environmental monitoring programs at the Argonne National Laboratory (ANL)[44] and at the Danish Atomic Energy Commission Research Establishment[45] are typical of the existing programs. The purposes of such programs, as stated in the ANL report, are to measure the natural radioactive content of the site and its environment, determine the magnitude and origin of any radioactivity above the natural levels, and, of primary interest, to detect any radioactive materials released to the environment by the particular operation.

At the ANL site the radioactive content of the environment was measured during 1958 by performing radiochemical analyses and total activity counting on more than 1300 samples of seven types, distributed as follows: water, 275; precipitation, 82; soil, 63; bottom silt, 146; plants, 57; filtered air, 689; and animals, 2. Most of the samples were of water, bottom silt in streams, and filtered air, because the most probable means of dispersal of radioactive contamination are water and air. Samples were collected at the ANL site and at locations approximately 10, 25, and 100 miles from the Laboratory. The sampling points, up to 25 miles distant, were located to indicate the extent and direction of contamination in the event that significant amounts of activity were released at ANL, and those at 100 miles were "reference" points intended for determinations of back-

ground levels of activity and for an over-all check for contamination of samples during collection, analysis, and storage. It is stated in the ANL report[44] that "this purpose has been realized for measurements of alpha activity, but because of the widespread occurrence of fallout from nuclear detonations these samples have served primarily to indicate the extent and magnitude of fallout activity over a large area." The presence of fallout activity in most samples has made it more difficult to interpret beta activities in terms of origin than alpha activities, but it was usually possible to distinguish between the beta activity from nuclear explosions, ANL operations, and natural sources.

The details of sampling locations and results of radioactivity measurements are given in the form of charts and tables. Since ANL waste water is discharged into Sawmill Creek, which, in turn, empties into Des Plaines River, special emphasis was placed on sampling these streams, including surface water and bottom silt. Radioactivity originating at Argonne was detected in water and bottom silt collected from Sawmill Creek below the ANL outfall. Compared with MPC's for the general population, the radioactive content of the water was 0.3 per cent of the MPC for uranium, 1.8 per cent of the MPC for plutonium, and approximately 3 per cent of the MPC for Sr^{90}. Activity added to the creek in ANL waste water was detected in bottom silt only within the first 20 yards of the bed below the outfall.

Except for this short distance in Sawmill Creek and a storage lagoon containing contaminated waste water, bottom silt from all locations contained only natural and fallout activities. With the exception of grass samples collected near a storage shed containing uranium, no evidence of activity originating at ANL was found in plants during 1958. The samples obtained near the storage shed contained alpha activity of about 4 $\mu\mu c/g$, as compared with normal concentrations elsewhere of 1 to 2 $\mu\mu c/g$. Surface soil collected near the storage shed contained up to five times the concentrations of alpha and beta activities normally found in the Chicago area, but the total activities in the other soil samples from all locations on and off the site were similar to those found earlier. Soil borings were collected near a plot of land formerly used by ANL as a burial ground for contaminated wastes, and normal activities were found. There was no evidence that any of the activity had moved from the plot to the surrounding area.

Fallout activity was not as great in soil samples as in other types of samples. It was greatest during the spring and fall and was particularly noticeable in air, precipitation, and plant samples. In 1958 the total long-lived beta activity in air-filter samples was approximately 3 $\mu\mu c/m^3$, and that in grass samples from "reference" locations averaged 253 $\mu\mu c/g$. With regard to fallout of radioactivity from the atmosphere, the report states that "from the results given it appears that grass tends to concentrate fallout activity and is thus a more sensitive indicator of fallout than soil, water, and possibly air, when only beta counting is used."

The annual report[45] on environmental radioactivity at the Danish establishment near Copenhagen describes a monitoring program that is even more detailed than the program at ANL, both in the types of samples collected and examined and in the frequency of sampling. Regular surveys were carried out on sea water, soil, air, bed soil from the fjord, eels, fresh water, rain water, grass, sea plants, and milk. Beta activity was determined, and in the case of grass, sea plants, and milk, the natural potassium was subtracted from the counting results. Also, strontium, cerium, cesium, and yttrium activities were determined in a number of samples, and alpha and gamma measurements were made occasionally.

The number of samples and frequency of sampling varied, depending upon the seasons and local conditions, but most samples were taken at least weekly from five to ten locations, with multiple sampling at each location. The procedures used in sample preparation and in the analyses are described in detail in the report. The results for each sampling period are given in tabular form. Gross beta results were treated statistically according to a system used in industrial quality control, and the statistical evaluations are summarized as appropriate.

Radiochemical analyses varied considerably, depending upon the type of sample and the significant radionuclides of greatest interest in the environmental medium represented. The radiochemical analyses were made of daily filtered air samples (combined to make a monthly composite), bimonthly grass samples, bimonthly samples of sea plants, and rain water collected after each rain. The analyses included determinations of one or more isotopes of strontium,

yttrium, cerium, cesium, ruthenium, rare earths, and praseodymium. In addition to the regular surveys, special surveys during the year included age determinations of fission products found in air and grass samples (which ranged from 50 to 400 days in air and 15 to 180 days in grass), analyses of activity in lichens and moss, which have large surface areas and slow rates of growth, and strontium analyses on the bone ash from three hares caught near the site.

After comparison of the data with those collected since 1956, when the program was begun, and with the MPC limits accepted for various environmental media, it was concluded that no increased radioactivity in the surroundings originated from the running of the Research Establishment. An environmental activity rise in 1958 was most evident in the air and grass samples, which showed approximately three times the fission-product activity found the year before. This rise was attributed to test explosions in 1958, as evidenced by the dates of sampling and the known dates of nuclear-test explosions. It was stated that the comparatively high Sr^{90} concentrations in grass and rain water suggested the need for close observation of these samples in the future. Of special interest was the increase of Sr^{90} in grass, since this would be likely to cause increased activity in the milk and consequently in human bones.[45]

(R. J. Morton)

Problems of Waste Disposal into Fresh-water Systems

Water has historically been used as a medium for disposal of industrial wastes. The nuclear industry, drawing heavily on previous industrial experience, has, as mentioned in the preceding review, employed the philosophy of using the environment in the disposal of wastes. Indeed, at many sites, both here and abroad, low-level radioactive wastes have been discharged to river systems. The complications involved in the dispersal of the radionuclides thus discharged are discussed here.

Concern about the effects of radioactive materials on river environments has existed since the first disposal of wastes from the Clinton Engineer Works (now the Oak Ridge National Laboratory) into the Clinch River on Mar. 6, 1944.[46] The question of whether any radioactive materials at all should be discharged to the environment has become highly controversial, not only in this country (for example, see Vol. 5 of the Congressional hearings on Industrial Radioactive Waste Disposal[28]), but also world-wide. At the International Scientific Conference on the Disposal of Radioactive Wastes held at Monaco in November 1959, delegates stated that it is inadvisable for any radioactive materials to be discharged to the environment. To the contrary, however, most of the major nuclear powers presented papers detailing the use of fresh-water systems as part of the dilute-and-disperse philosophy of waste disposal or have so indicated in the past.

The British discharge low-level liquid wastes from Harwell, Aldermaston, and Amersham to the Thames River above the water-intake pipes for London's drinking water supply.[47] The French stated that "the elimination by dilution of slightly radioactive effluents is practically inevitable."[48] The wastes from Saclay, Fontenay-aux-Roses, Le Bouchet, and Marcoule are discharged to the sewers and eventually to the rivers or directly to the rivers.[49] The wastes from Chalk River find their way into the Ottawa River.[50] Practically all the research centers and reactor stations in the United States discharge some radioactive materials to the environment. Of all the low-level activity discharged to the environment in the United States since 1944 without further control, Hanford has released 96.4 per cent and Oak Ridge has released 3.4 per cent.[51] These percentages are large, but it must be remembered that these installations have been in operation much longer than any other installations in this country. It should also be pointed out that the released activity has consisted essentially of short-lived nuclides in concentrations below MPC values.

The Russian technical literature indicates that some methods of disposal to the environment are being evaluated, presumably for present or proposed usage. Peredel'skiĭ and Bogatyrev[52] have investigated the hazards to surrounding areas posed by insects leaving radioactively contaminated basins. Agafanov[53] and Timofeeva-Resovskaya[54] have shown that radioactive material is rapidly extracted in a series of fresh-water plankton, water plants, sands, and clays.

In the early days, discharges to fresh-water systems were regulated solely by compliance with ICRP or NCRP recommendations of maxi-

mum permissible concentrations in drinking water or by use of some fraction of this concentration. It was realized even then, however, that some of the nuclides could be reconcentrated in the fresh-water system. Today it is almost universally recognized that the limits on discharge to fresh-water systems must be set so that the dosage received from all sources is less than that recommended by the ICRP and NCRP. This requires a study of each discharge site and of the use made of the receiving water.

The receiving stream may be used for several purposes, for example, drinking, bathing, boating, irrigation, livestock watering, fishing, aquatic plant cultivation, and industrial processes; and it is, of course, the natural habitat of various species of wildlife and a breeding place for many insects. Therefore the exposures resulting from each use must be evaluated. If it is assumed that a certain percentage of the lifetime dose can be received from this phase of the environment, the quantity and makeup of the radioactivity which may be discharged to the stream can be computed. The calculations are complicated, however, because the concentration of radioactive material in the water system is being continually reduced by dilution, diffusion, and radioactive decay. Further, the various isotopes are selectively concentrated in the sediments and by the biota in the river, and they move into the food chain with various degrees of discrimination, depending upon the composition and metabolism of the receiving body.

Since each site must be studied individually, papers on how to conduct such a study must be general. Davis[55] details the probable source of wastes to streams and points out that the chemical and physical properties of the wastes, the limnological factors, and the biota can significantly affect their dispersion. The Columbia River below Hanford is used as an example, but no over-all integration of the mechanisms involved in dispersing the wastes is attempted. Tsivoglou et al.[56,57] describe in general terms the mechanics of making a stream survey. They state that the major controls should be water, biological, and sediment sampling and that the extent of sampling should be determined by the purpose of the survey. Analyses of water give instantaneous quantitative determinations of radionuclide concentration, whereas sediments and biological samples give a measure of the selective accumulation of radionuclides. Because of the integrating effect of the biota and sediments, analyses of them do not give an indication of the time during which the activity is accumulated or the initial concentration of the activity in the water.

In an investigation of the Columbia River below Hanford,[58,59] the permissible concentration for P^{32} in water was determined on the basis of the NCRP standards for occupational and population-at-large exposure as related to the human consumption of fish and the prevention of somatic damage to the fish. The maximum observed concentration of P^{32} in the fishes in the Columbia River was about 100,000 times that in the water. Therefore the consumption of fish is the limiting factor in the disposal of the induced activity to the Columbia River. It is stated that "the concentration of radioactive substances will vary between species and tissues and will fluctuate according to food habits, life cycle, and seasonal changes."[58] For example, in the Columbia River, Sr^{90} has been found to be concentrated up to 10,000 times in algae, 1000 times in insect larvae, and 1000 times in fish. In the Columbia River fish, the concentration of P^{32} in the feces is about six times that in the fat. The major accumulations of P^{32} are in the feces, scales, bone, and liver. Further, the activity levels in the fish are 75 times higher in the late summer than in the winter. This is believed to be due to the increased water discharge and decreased concentration in the water during winter.

With the accumulation of data have come attempts to devise mathematical models to describe the fate of isotopes in river systems. Because of the complexities, including radioactive decay, dilution, diffusion, and selective uptake by sediments and biota, only simplified models have been attempted. The reviewer has determined the concentration downstream from the point of injection by analytically solving the diffusion equation for simple geometry and velocity transport.[60] In this model it was assumed that all the mechanisms causing reduction in the radioactivity of the water, except radioactive decay, could be fitted by an exponential curve.

Harris[61] has shown by simple statistical methods that it is possible to obtain information on the removal of radioactive materials from water by an examination of the water samples alone. Measurements of the sediments and biota do not, of course, provide this information un-

less the total weight of the biota and sediments involved in the uptake is known. The statistical method of Harris is based upon an early Columbia River survey.[62] A comparison is made of the decay curve at the point of plant effluent, where it is assumed that the relative concentrations of nuclides do not change with the water concentration, and decay curves at downstream points. The differences between the predicted concentrations and actual concentrations are assumed to be due to removal by sediments or the biota. Selective uptake is determined by comparing the decay curves of water samples taken at downstream stations with the decay curve of the point of plant effluent. The practical application of this method is limited since it is applicable primarily to effluents containing short-lived nuclides (i.e., half lives less than the flow time) and since it assumes that the relative concentration of nuclides in the effluent remains fairly constant. Both these conditions would be true for induced activity in reactor cooling water but would be unlikely to be true for most other radioactive waste streams.

The importance of selective removal of radionuclides should not be minimized. Krumholz and Foster[63] state that "each organism in each environment has specific requirements for the different chemical elements." To date no satisfactory mathematical model for analyzing these situations has been developed. It seems unlikely that such a model can be evolved until more basic information is obtained on the mechanism and rate of eddy diffusion, isotope uptake by suspended and bottom sediments, and isotope uptake by the biota. (F. L. Parker)

Handling of Liquid Wastes at U. S. Reactor Installations

A review of the practices employed in the handling of the liquid wastes from U. S. power reactors indicates that the general philosophy is the same for nearly all, i.e., store high-level wastes, treat and reuse the low-level wastes whenever possible, and discharge the low-level wastes that cannot be reused. For this discussion, the following arbitrary definitions are adopted: a low-level waste is one which, after dilution by the condenser cooling water, has a sufficiently low activity concentration to be discharged to the environment; all other radioactive wastes are considered to be high-level wastes. Most reactor high-level wastes contain far less activity than high-level wastes from a spent-fuel reprocessing plant. Hazards reports on the individual reactors[2–20] reveal differences in techniques to suit local conditions and preferences of designers. The waste-handling systems are designed, of course, for the particular waste solutions existing at each site. The practices at various sites are summarized in Table V-1.

Sources of Liquid Wastes

For the most part, no explanation of the waste sources listed in Table V-1 is required. A brief discussion of some of the sources is warranted, however.*

The principal source of liquid waste for the water-cooled reactors is the primary coolant, which becomes contaminated by activated corrosion products and by fission products from ruptured fuel elements. In all these reactors the primary coolant is continuously purified by ion exchange, and therefore the coolant does not represent an actual liquid waste.

Water used in spent-fuel handling and in storage basins is subject to contamination by fission and corrosion products. For example, the fuel removed from sodium-cooled reactors is washed to remove sodium, and the water from this source is contaminated with Na^{24}.

A number of the reactor installations employ ion exchangers for treatment not only of the primary coolant but of other radioactive liquid wastes. Regeneration of the ion-exchange resin results in solutions containing radioactive materials, as well as the regeneration chemicals. Not all resins are regenerated, however. In particular, resins used for primary coolant purification are discarded when they are spent. Usually, the spent resin is sluiced with water to tanks in which it is stored. The sluice water is then a low-level liquid waste. In general, a spent resin is regenerated if the regeneration process will yield a low-level waste, and it is not regenerated when a high-level waste would result. In the latter case, it is advantageous to store the activity in the form of the resin. In small installations, it may not be worthwhile to regenerate under either circumstance.

The heavy water used in the CVTR and the FWCNG reactors is a potential waste that is too

*It should be pointed out that, of the reactors discussed here, some have already been completed but others are still in the planning or building stages.

Table V-1 SUMMARY OF LIQUID-WASTE-HANDLING PRACTICES AT U. S. POWER-REACTOR INSTALLATIONS*

	Yankee	Shippingport	SM-1 (APPR)	Indian Point	Saxton	N.S. *Savannah*	BONUS	Dresden	Pathfinder	Humboldt Bay	Vallecitos	Elk River	Piqua	HRT	CVTR	FWCNG	EGCR	Enrico Fermi	Hallam
Reactor type†	PWR	PWR	PWR	PWR	PWR	PWR	BWR	BWR	BWR	BWR	BWR	BWR	OMR	AHR	PTR	GCR	GCR	FBR	SGR
Nuclear heat rating, Mw	485	230	10	585	20	69	50	630	203	163	20	58	46	10	60	153	84	300	240
Coolant	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	Organic	D_2O	D_2O	CO_2	He	Na	Na
Sources of liquid waste:																			
Primary reactor cooling water	x	x	x	x	x	x	x	x	x	x	x	x							
Fuel handling and storage basin	x	x	x	x	x		x	x	x	x	x	x	x		x	x	x	x	x
Radiochemical laboratory	x	x	x	x	x	x		x	x	x	x	x		x	x		x		
Laundry		x		x														x	x
Equipment and tool decontamination	x	x			x			x	x	x		x		x	x	x	x		x
Boron solution for reactor shutdown	x	x		x	x		x	x		x		x							
Demineralizer regeneration and sluicing	x	x	x	x	x	x	x	x	x	x	x	x					x		
Waste-treatment facilities:																			
Ion exchange (in addition to primary coolant demineralizer)	x	x		x					x	x	x	x	x		x	x	x	x	
Concentration	x	x					x	x	x			x		x					
High-level liquid waste handling:																			
Permanent storage of liquid on site					x			x		x									x
Liquid shipped for storage off site			x	x		x			x		x	x	x	x			x		
Liquid converted to solid form	x	x					x												
Low-level liquid waste handling:																			
Stored for disposal off site						x													
Discharged to environment:																			
—Stream, lake, or ocean	x	x		x	x		x	x	x	x	x	x	x	x	x		x	x	
—Seepage pit														x		x	x		
—Sanitary sewage leaching field																			x
Hazards-report reference	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20

*For more precise identification and location of most of these reactors, see USAEC Report TID-8200(1st Rev.), which is available free from the AEC Office of Technical Information Extension, P. O. Box 62, Oak Ridge, Tenn.

†The entries on this line indicate the following: PWR = pressurized-water reactor, BWR = boiling-water reactor, OMR = organic-moderated reactor, AHR = aqueous homogeneous reactor, PTR = pressurized-tube reactor, GCR = gas-cooled reactor, FBR = fast-breeder reactor, and SGR = sodium-cooled graphite-moderated reactor.

expensive to be discarded or stored indefinitely, and therefore the radioactivity is kept to a low level by ion exchange. If the heavy water became contaminated with light water, it would be sent to a heavy-water plant for recovery.

Waste-treatment Facilities

The major waste-treatment facilities provided at the various reactor stations are also indicated in Table V-1. Facilities common to all, such as tanks and monitoring equipment, are not shown.

As indicated, a number of reactors employ ion-exchange treatment of waste in order to make it suitable for reuse or to reduce the activity content to a level suitable for release to the environment. Some facilities use ion exchangers other than those for primary coolant purification, and some use the primary coolant ion exchangers for treatment of compatible wastes as well, for example, Yankee and Indian Point.

Waste liquids such as the water used in fuel-storage basins are of high purity even when contaminated with radioactive fission and corrosion products, and it is often advantageous to treat this water rather than to discard it. At least two of the reactors (Saxton and Elk River) that use boronated water in reactor shutdown employ ion exchangers to remove the boron. Interestingly, the facility that employs boronated water in the spent-fuel storage basin (Elk River) uses an ion-exchange resin that leaves the boron in the water. Vallecitos employs ion exchange to make the waste suitable for discharge to the environment because it is (or was at the time of preparation of the hazards report) permitted to discharge only background-level wastes.

Concentration facilities are provided at seven reactor stations, as indicated, to reduce the volume of high-level waste to be stored. At two stations, Yankee and Shippingport, the concentrated liquid is mixed with cement in drums. At BONUS, the high-level waste liquid will be evaporated to a slurry which solidifies on cooling.

Handling of High-level Wastes

The bulk of the activity contained in high-level liquid wastes is stored in either liquid or solid form. Relatively few stations have facilities for long-term storage of liquids at the reactor site. Nine stations ship liquids to storage facilities elsewhere. Four stations concentrate liquids before storing them. Three stations convert high-level wastes to solid form, and at least one of these (Shippingport) plans to use ocean burial of the drummed solids.

Handling of Low-level Wastes

All reactors except the N.S. *Savannah* and the SM-1 (formerly APPR-1) discard low-level wastes to the environment. The SM-1 facility has only one kind of waste, i.e., so-called "hot" waste. The "environment," in most cases, is a nearby stream. The Humboldt Bay plant is to be located on an arm of the Pacific Ocean; the Enrico Fermi station is located on Lake Erie. Two plants, Yankee and Dresden, discharge to impounded streams; the EGCR will be doing so if a proposed dam is built.

Three of the reactors discharge low-level wastes to unlined earth pits. At Oak Ridge these pits are known as seepage pits. Water seeps down into the Conasauga-shale formation which underlies much of the area. Most of the activity in the water becomes fixed on the shale before the water finds its way to ground water or to a stream. The FWCNG reactor would discharge to ponds which once were used in phosphate mining operations. These ponds would be large and would serve both as the source and receiver of condenser cooling water. All the waste from FWCNG would be passed through demineralizers before being discharged.

Estimates on the amount of activity released to the environment in liquid form have been reported for two of the reactors. It is estimated that the Dresden[9] facility will release about 10 curies per year and that the Yankee[2] facility will release about 0.001 curie per year.

(W. L. Albrecht)

Fission-product Release Tests

The fractional release of fission products from melted fuel elements and the meteorological factors governing their subsequent dispersion are pertinent to hazards evaluation. A number of field release tests have been conducted at the National Reactor Testing Station (NRTS) and are described in two recent reports.[64,65] Of particular interest in these tests were the fractions of the various isotopes released under given conditions, the physical form of the vari-

ous activities, and the dispersion of this activity by the atmosphere.

Operation Boot, conducted at NRTS[64] in May 1958 by the General Electric Company's Aircraft Nuclear Propulsion Department, was planned as an integrated field experiment for determining the pertinent factors in the release and dissemination of activity. A metallic fuel element was partially melted by a 30-sec pulse of fission heat in a restricted flow of air, and the air was discharged at an effective stack height of about 210 ft. Three independent estimates of the fraction of I^{131} released gave the following values:

Estimating method used	Release fraction
Residue analysis	0.25
Effluent sample analysis	0.026
Source strength to give measured isopleth concentrations	0.025

The intrinsic uncertainties associated with these evaluation methods precluded resolution of the factor-of-10 variation obtained. No one method was sufficiently definitive that the results of the other methods could be discounted. The random samples of residue that were analyzed may not have been representative. The effluent sampling was not isokinetic and may have suffered from preferential entrapment of gases or particulate matter. On the other hand, it was not believed to be justifiable to employ the measured atmospheric concentration to determine the release fraction since this was one of the parameters under investigation.

Some release of Sr^{91} (estimated fraction, 0.006) was observed in the effluent samples but was not confirmed because of the absence of chemical separation of downwind samples in which, if present, it was masked by other isotopes. The fallout radioactivity observed on vegetation and objects was too small for quantitative determination of individual isotopes. Particle-size data on samples from the exit duct showed unusual numbers of large and very small particles, with relatively few particles of in-between sizes. However, the cascade impactor used had not been calibrated for the conditions encountered in the test, and therefore the size distribution had to be based on an assumed density of 9.0 g/cm^3 and the calibration supplied by the manufacturer. The released I^{131} was apparently not in particulate form in the duct to the charcoal traps in which iodine was found. Presumably the iodine was in particulate form or associated with particulates when it was collected by the downwind air samplers. The sampling efficiency of these high-volume air samplers is not known. A large discrepancy (factor of 700) was observed between the amount of iodine collected in downwind charcoal traps and that collected by the high-volume air samplers. The data from the traps were discounted on the basis of requiring a release of more than the total inventory, although it is not obvious that this was justifiable.

Although the system used for this test was satisfactory for the planned study of fuel-element damage and it provided a source of released activity, the instrumentation does not appear to have been sufficiently adequate to determine accurately the release values under conditions where the entire release occurred during a period of seconds. The field measurements indicate that the shape of the isopleth pattern may be described by Sutton's diffusion theory, but, because the source strength was not established, the absolute values of the isopleths may be incorrect. For the same reason it is not possible to determine quantitatively the degree to which Sutton's equations describe the distribution of air-borne material during conditions of strong lapse (condition of greatest atmospheric mixing and thus greatest dispersal of activity) for the distances considered. It was recognized that the measurements taken during the diffusion of a cloud of fission products were not precise and that the analyses of the data were extremely complex. For this reason a neutral or weak lapse condition was originally selected. It is unfortunate that, for permission to conduct this release, the requirement of a strong lapse condition was imposed at the last minute.

A subsequent series of nine fission-product field release tests (designated *A* through *I*) were conducted at NRTS[65] during the period July to September 1958 by the Nuclear Aircraft Research Facility (Convair). The objectives of these tests were to reduce the uncertainties in hazard-analysis assumptions with respect to release percentages, particle size and size distribution, and atmospheric diffusion and deposition and to obtain direct evidence of the downwind hazard resulting from fuel-element

meltdown. Both aged and freshly irradiated (green) metallic fuel-element specimens were melted in an induction furnace. The furnace was designed for rapid melting of the specimens (in about 2 min). After the specimens melted, the temperature was held at about 100°C above the melting temperature for 10 min, and then heating was stopped. In two of the tests (*D* and *H*), thermocouple failure led to excessive temperatures that were estimated to be 400 to 800°C above the melting temperature. The furnace was purged with a metered air flow of 1 cfm, and the effluent was released directly to the environment 5 ft above ground level. Evacuated flasks were used to sample the furnace effluent at intervals during melting, and an additional flask was used to collect an integrated sample. The meteorological condition selected for five of the releases was moderate inversion (condition of least atmospheric mixing and thus least dispersal of activity) and for the other four moderate lapse. In the case of release *A*, the inversion did not develop from a weak lapse until about the time melting began.

Analyses of the sequential samples showed sequential release of different fission products. Essentially all the iodine was released within the first 2 or 3 min, followed by cesium, strontium, ruthenium, zirconium-niobium, and cerium, in that order. Essentially all the activity released was liberated within a period of 6 to 8 min. Under the conditions of sustained melting, the release fractions observed were, approximately,

Iodine	0.50
Cesium	0.30
Strontium	0.01
Ruthenium	0.008
Cerium and zirconium-niobium	Small

Almost all (instead of the predicted value of one-fourth) the cesium and strontium appeared as aerosols rather than more gross particulates. The agreement of the release percentages obtained by various methods was rather good. The methods used included measurements of pre- and postmelting gamma-ray spectra and gross gamma-ray dose rates from aged fuel elements, pre- and postmelting radiochemical analyses of green fuel elements, fractional sampling of the effluent at the furnace, and network sampling of the effluent cloud.

The observed centerline air concentrations during lapse conditions were within a factor of 2 of the values predicted from typical lapse parameters, and the measured centerline concentrations during inversions were an order of magnitude less than predicted using typical inversion parameters. These lower concentrations are considered to be attributable to better vertical diffusion. An increase in the value of the typical vertical diffusion coefficient by about a factor of 10 (to 0.4) is recommended. The typical values referred to are listed in Table V-2.

Table V-2 TYPICAL ATMOSPHERIC DIFFUSION PARAMETERS

	Symbol	Units	Value during lapse condition	Value during inversion condition
Stability parameter	n	None	0.25	0.55
Diffusion coefficient	C_y	Meters$^{n/2}$	0.40	0.40
Diffusion coefficient	C_z	Meters$^{n/2}$	0.40	0.05
Mean wind speed	u	Meters/sec	5.0	3.0
Length of source	h	Meters	0	0
Deposition velocity	V_g	Cm/sec		
Iodine			2.5	0.25
Other fission products			1.0	0.1

It was concluded that additional data would be required to resolve the variations in deposition velocity (V_g) from the predicted values. The average measured proportionality constants between cloud and ground contamination were nearly the same as those used in predictions for release during lapse conditions. However, the measured deposition velocities for inversion and lapse conditions were not significantly different from each other. The value for iodine was not significantly different from that for bone seekers, but the cesium deposition rates were an order of magnitude lower than for other isotopes.

The calculated fission-product inventories were considered to be in good agreement with analytical results. Because of the approximations, such as in estimating average breathing rates, the lower respiratory tract deposition values found in the rats and dogs were not con-

sidered to be significantly different from the deposition values calculated for the NCRP standard man. As expected, the internal dose (iodine, thyroid) that would result from breathing while submerged in the green fuel-element release cloud was measured to be about three orders of magnitude greater than the external dose. The measured external gamma-ray dose under lapse conditions (at 400 meters) was about one-half that predicted. No more than 5 per cent of any aerosol, except iodine, was found to penetrate the pleated filter of the high-volume air sampler. However, it was found that about 25 to 45 per cent of the iodine, based on total iodine collected by both the pleated filter and backup carbon cartridge, had penetrated the pleated filter. This is attributed to the interaction of the local dust and the iodine and depended upon the concentration of the dust. As a result, more penetration was observed close in than at farther distances downwind.

Tests such as these are valuable, not only for the additional information they provide on release fractions and diffusion under particular field conditions, but also to point out areas, such as deposition velocity, in which further study is needed. (T. J. Burnett)

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Activity Releases at Oak Ridge National Laboratory

During the period between Oct. 28 and Nov. 20, 1959, three incidents involving the accidental release of activity occurred at ORNL. The first incident occurred during the period between October 28 and November 6 when high-level radioactive waste was released into the process-water waste system. The discharge of approximately 55 curies of Ru^{106} and Ce^{144} into White Oak Creek resulted. The second uncontrolled release of activity occurred on November 11 and 12, at which time about 15 curies of Ru^{106} was discharged from one of the stacks that handles gaseous wastes. The third of the incidents involved an explosion on November 20 in the Thorex Pilot Plant which released between 1 and 5 g of plutonium into the building. A report (as yet unpublished) by the ORNL investigating committee (C. E. Winters, Chairman; K. A. Kraus; and A. C. Upton) describes the events leading up to the accident and makes several recommendations to prevent the recurrence of such an accident. Of the plutonium released, about 0.6 g was spread into the adjacent streets and onto the walls of the nearby buildings. A discussion of this incident was presented[1] in *Nuclear Safety*, Vol. 1, No. 3.

Activity Release to Waste System and White Oak Creek

Various operations at ORNL result in considerable amounts of radioactive waste which must be disposed of without creating an undue hazard. The methods used for disposing of the solid, gaseous, and liquid wastes depend upon the activity level.[2] Mildly contaminated waste process water (0.1 to 1.0 μc/gal), amounting to about 700,000 gal/day, makes up most of the material to be handled by the waste-disposal system. In describing the waste-disposal system, Browder[2] states that:

> The sources of the waste process water are equipment cooling systems, floor drains, decontamination pad drains, storage canals, laboratory sinks, and discharges from low-activity operations. The waste is collected, sampled, and discharged to White Oak Creek continuously. Much of the radioactive contamination put through this system is a result of equipment failure, human error, or accidents that cause a misdirection of contamination from the highly active chemical waste system.

Waste process water from the various sources is collected and routed through a network of clay pipes to a diversion box near the inlet to the settling basin, as shown in Fig. 1. The water is continuously monitored for activity at the diversion box, and every 4 hr samples are collected and analyzed for gross beta activity. The waste collection system is divided into several sections, and each section contains a station (i.e., a concrete manhole) at which flow rates can be measured and samples can be obtained for activity analyses. When the activity level at the diversion box rises to about 150 counts/min per milliliter of water or greater, the source responsible for the increase is located by checking activity levels at the concrete manholes. Corrective action is either taken at the source or the waste water is diverted to the process-waste-treatment plant. A more complete description of the system, including flow sheets, has been given by Browder.[2]

During the period between Oct. 28 and Nov. 6, 1959, there was an abnormal influx of excessively high activity to the waste system.[3] An abnormally high activity level, about 12,000 counts/min/ml, was first noted in the diversion box on October 28. It was known that some activity had been discharged into the process-waste system from operations in the radioisotopes production area, and it was therefore initially assumed that this discharge had caused the high activity readings in the diversion box. The source of the activity was not checked. On the following day (October 29), the activity level in the diversion box dropped; but, on Octo-

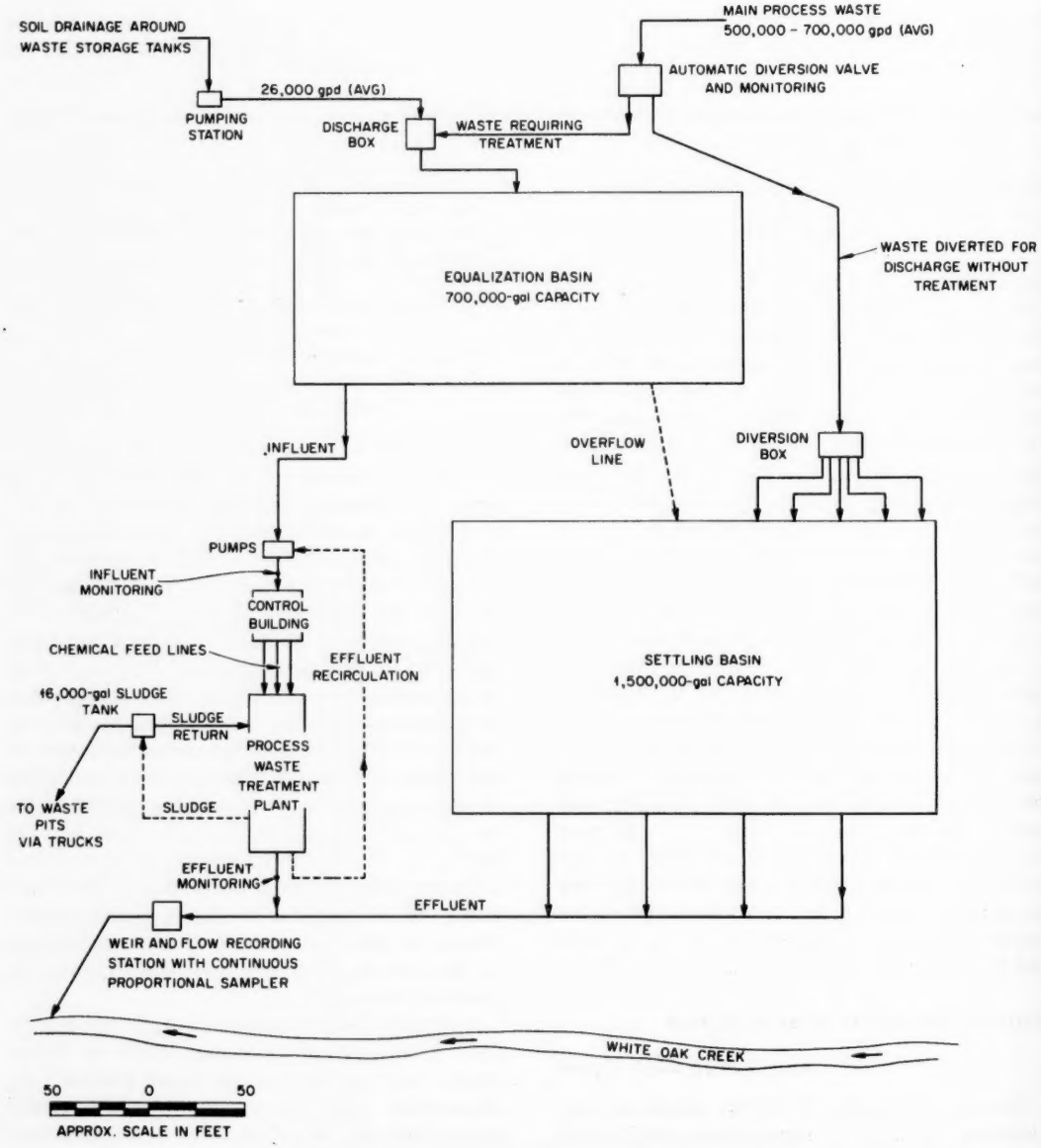

Figure 1—Flow diagram of ORNL process-waste system.

ber 30, samples taken at the manhole in the discharge line from the Thorex Pilot Plant (Building 3019) indicated that activity was possibly being discharged there.

Operations in Building 3019 during this period consisted of the processing of MTR plutonium fuel assemblies. The products being recovered were the rare-earth isotopes Cm^{242} and Am^{242}. The various separation processes had been completed, leaving the rare earths Cm^{242} and Am^{242} as products in the evaporator to be concentrated. On October 31, the evaporation process was approximately 50 per cent complete when high activity was detected in the settling basin. Under normal conditions, evaporation is accomplished by maintaining continuous steam

pressure on the heat exchanger. Leaks occurring under these conditions would be into the product rather than into the condensate.

On October 31, a sample from the manhole in the discharge line from Building 3019 showed 2000 counts/min/ml, whereas the diversion box material, after dilution, showed 11,000 counts/min/ml. The condensers and coolers in Building 3019 which were not absolutely necessary for operation were thereupon shut off to reduce the volume of waste discharged to the settling basin, but it was still not known whether Building 3019 was the source of the high activity level in the diversion box. The samples from the manhole were therefore analyzed to determine the main components of the activity. The analysis showed that the isotopes present matched those of the material being processed in Building 3019 and that the previously reported activity measurement of 2000 counts/min/ml was in error, a more nearly correct reading being 91,000 counts/min/ml.

All operations in Building 3019 were then shut down in order to reduce the flow to the settling basin, which was about to overflow into White Oak Creek, but activity continued to be discharged from the building because the flow of cooling water to the equipment was not turned off. A leak test of the evaporator on November 1 indicated that leaks existed in both the steam trap and the heat exchanger.

A pump was therefore installed at the manhole in the discharge line from Building 3019 to intercept the waste flow and to transfer the waste to the waste-treatment system. This scheme was only partially successful because, as was determined three days later, operation of the pump was unreliable in that it would stop for short periods of time, possibly as a result of the liquid bypassing the weir where the pump was located. The activity did not drop to a normal level until all flow of process water into Building 3019 was shut off. Later, when the process water was turned back on for laboratory use in Building 3019, the activity level again increased and remained high until Building 3019 operations were again discontinued and the process water was shut off.

On November 1, the gross beta activity at White Oak Dam reached 94 counts/min/ml (mainly Ru^{106} and Ce^{144}), and the dam was closed. On November 3, controlled release to the Clinch River was begun; and, by November 6, the lake formed by White Oak Creek when the dam was closed had been completely drained. The maximum contamination occurred just below the mouth of White Oak Creek and was 3.7×10^{-6} $\mu c/cm^3$. This concentration is below the maximum permissible concentration in water of 7×10^{-6} $\mu c/cm^3$. The total activity release was 55 curies.

This activity release did not create a serious problem outside the Laboratory since it was adequately handled by the waste-disposal system, but the incident pointed out several weak points in the method of handling waste process water. Because of the nature of the accident, no personnel exposure resulted. The Radioactive Operations Review Committee at ORNL, in its report of the incident, suggested the following:

1. Reliable, low-background monitors should be installed as close as possible to each source of process waste. The flow rate and the activity level should be continuously recorded both at the source and at a central station.

2. The Waste Treatment Plant should routinely treat *all* process waste. To make this possible, large users must reduce their effluent to the process-waste system. Large-volume flows, such as condenser cooling water, with very low probability of contamination should be either recirculated or held up, monitored, and released to storm sewers.

3. Means should be provided for handling very high levels of activity in the waste system. Pumps and storage capacity should be provided either for recycling of the Waste Treatment Plant effluent back through the plant for a second pass or for pumping the waste directly to the high-level-waste pits.

Activity Release from Stack

The gaseous wastes from the X-10 area are collected and sent to one of the several stacks. In order to control the amount of activity released to the atmosphere, radioactive solids, as well as chemically reactive gases such as iodine, are removed by filters, electrostatic precipitation, or adsorption equipment.[2] On November 11 and 12, activity was released from the gaseous-waste system in quantities that did not constitute a serious hazard to personnel but which, at the same time, indicated that serious weaknesses existed in the gaseous-waste-disposal system.[4]

The physical layout of the gaseous-waste-disposal system associated with the brick stack from which the activity was released is shown

in Fig. 2. As may be seen, the gaseous wastes flow through an electrostatic precipitator and a filter before entering the stack. Therefore, if these pieces of equipment are in place and operating as specified in the design of the facility, the amount of particulate activity at the fans should be negligible, and the bulk of the activity being released from the stack would be due to xenon or krypton isotopes.

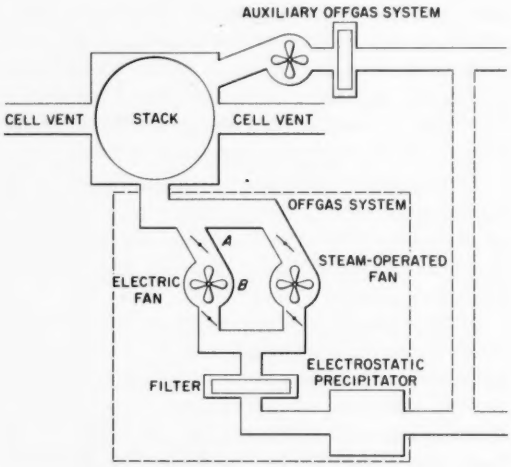

Figure 2—Typical gaseous-waste-disposal system at ORNL.

Immediately prior to the release of activity on November 11 and 12, maintenance work on components in the system was required on two occasions. The exhaust damper of the electric fan (indicated by A in Fig. 2) did not close tightly, and therefore routine repairs were started. Following the completion of this maintenance, the fan operated for about 4 hr, and then it was shut down because of a coincidental bearing failure. In the course of performing maintenance on the bearing of the off-gas fan, the fan housing interior and impeller were partially decontaminated with weak caustic and weak nitric acid to reduce the surface dose rates at the equipment from 80 r/hr to about 5 r/hr. The repairs were completed on November 11, and at 3:30 p.m. the electric fan was turned on for approximately 30 min of initial operation. At 4:00 p.m. the electric fan was turned off, and the steam-operated fan began operation since the normal procedure following maintenance on bearings was to commence operation at the beginning of the day so that the bearing performance could be watched during the first day of operation. Therefore the continuous operation of the electric fan was scheduled for the following morning.

Contamination was first detected in the isotope production area at 4:00 p.m. on November 11. During the next two hours (4:00 to 6:00 p.m.), efforts were made to determine the extent of the contamination. By midnight it was established that the contamination was due to ruthenium. The period from midnight to 8:00 a.m. was spent in cleaning up the areas of contamination. By 8:00 a.m. the extent of the contamination was well known, and it was suspected to be due to fallout from the brick stack, but it was not determined as to when the release happened.

At 8:15 a.m. on November 12, the electric fan was put into operation as scheduled. Upon realizing that the electric fan was suspected to be the source of contamination, it was turned off, and the steam fan was turned back on. By this time, however, the electric fan had operated for 20 to 30 min. Between 9:30 and 10:30 a.m., new fallout was discovered; and therefore, at about noon, orders were given to close down as many processing operations as possible that used the off-gas facilities of the brick stack. Following this, the steam-operated fan was closed down, and an auxiliary fan and bypass off-gas connection were put into operation.

As a result of the activity release from the stack, contamination was spread over portions of the Laboratory area. The first release spread contamination in the east-west direction, the highest readings being in the isotope production area where a rough average of readings taken with a G-M survey meter at grass-top level indicated about 20 mr/hr. The dose rate in the east parking lot was about 1 mr/hr. The second release spread contamination predominantly in the northwest direction. During this fallout, readings as high as 50 mr/hr were indicated within 300 ft of the stack base.

General washing of sidewalks and streets resulted in reducing the levels to those which existed prior to the release. A number of people had contamination on their shoes, but preliminary results from urinalyses of samples from about 50 persons indicated no ruthenium or strontium activity.

In the course of performing maintenance on the exhaust system, it is probable that both vibration and the decontamination procedure loosened scale from equipment which contained activity.

Subsequently the loosened activity was discharged up the stack. Although this may have been the direct cause of the incident, the mere fact that activity in large quantities was available to be released casts suspicion on the adequacy of the gaseous-waste-disposal system. Although electrostatic precipitators and filters are provided to control the solids capable of being exhausted, the fact that the fans were highly contaminated indicates either that the equipment was not functioning as intended or that there were periods of operation when the equipment was not in place. Furthermore, although a monitor is provided to detect activity in the stack, the system is not capable of detecting particulate activity which deposits out on the walls of the duct ahead of a collection filter.

This incident, as well as the preceding one, points out the dilemma that exists at all nuclear facilities with regard to the proper balance of economy and safety. Since absolute safety can never be accomplished, to what degree should safety dictate both design and operation of a facility? Also, how much can one afford to pay for additional safety? Both these incidents support the contention that the design and operation of all nuclear facilities should be based on adequate safety for all probable events. The philosophy of safety followed in the design and operation of reactors has resulted in a very minimum of incidents, and it appears that, in designing and operating facilities other than reactors, greater emphasis should be placed on safety for the maximum credible accident.

Actions Taken by ORNL Following Incidents

Following the recent activity releases at ORNL, several important actions have been taken to prevent recurrence of the recent incidents and to minimize the consequences of any credible accidents that would lead to an activity release. For example, new regulations and standards have been issued for all operations involving radioactivity.[5] All high-level radiochemical operations have been closed down until all the hazards can be reevaluated. This automatically includes a ban on the use of unknown and unevaluated reagents in processing operations. Work has been initiated which will lead to a more quantitative relation between smear counts, direct alpha-monitor readings, and actual surface contamination. Action has been taken to form a plant squad of auxiliary health physics surveyors on rotating shifts.

Further, three new review committees have been established to assess the hazards involved in various operations at ORNL. These new committees (Radioactive Waste Disposal, Radiochemical Plants, and Hot Cells and Sources) replace the former Radioactive Operations Review Committee.

Of these actions, the most significant concerns the general criteria for containment of radioactive operations. The new regulations specify that[5] "the general criteria which have been adopted for containment or confinement of reactors will apply to radioactive operations at ORNL, including radiochemical pilot plants, 'hot' laboratories, and experimental setups, as well as to the ORNL reactors." Therefore the following criteria, which had not been specified in the past, must be satisfied in the future:[5]

> The maximum credible accident must be contained or confined to the degree necessary to preclude discharge into the Laboratory area and its environment of concentrations or amounts of radioactive materials which are injurious to health or which will interfere with other Laboratory programs.
>
> Two lines of defense must be present to prevent the escape of radioactive materials from our operations via waste streams.

In order to assure that two lines of defense exist at all times, adequate monitors must be provided to indicate the failure of either line of defense.

For gaseous wastes, each installation, or group of installations, must provide its own gas-cleaning system. The large stacks will not be used as a means for discharging or disposing of solids, but the cleaning facilities at the large stacks may serve as a second line of defense.

For liquid wastes, each installation, or group of installations, must provide means for preventing discharge of excessive amounts of radioactive materials into the laboratory waste system. The retention basin, settling basin, and waste-treatment plant will serve only as the second line of defense.

The new regulations state that:[5]

> These criteria will apply not only to future operations, but also to those in existence. A formal hazards report will be required for those operations involving >1 g of plutonium (or equivalent of other

alpha emitters or isotopes of comparable hazards) or >1000 curies of beta and gamma emitters. The reports will be submitted to the appropriate ORNL review committee and must be approved by the committee and ORNL Management prior to initiation of operation.

The new safety criteria adopted at ORNL are of interest to radiochemical plant designers and operators. For many years the hazards of operating reactors have been well recognized, and therefore the safety criteria generally applied to reactors have resulted in designs which tend to minimize accidents or which are capable of confining the activity release to an acceptable level. The AEC in its procedure for licensing the construction and operation of reactors requires that the construction and later the operation assure adequate safety.

The editors believe that the hazards of high-level radiochemical plants warrant similar attention before construction and periodically during operation. In the March 1960 issue of *Nuclear Safety*, we suggested that a fresh look at the hazards associated with existing facilities is in order, especially older ones wherein the containment provisions might not have kept pace with the operations carried out in them. The need for continuing review is substantiated by the fact that, of the unplanned activity releases in recent years, most have been from radiochemical facilities rather than reactors.

The new regulations adopted at ORNL will undoubtedly buy additional safety, but the cost of this safety is yet to be determined. At a time when the demand in the nuclear industry is for a reduction in costs for producing nuclear power, the purchase of additional safety at the expense of economical nuclear power will no doubt be received unfavorably. However, it must be remembered that those areas of the nuclear industry which are still in the developmental stage must necessarily approach the problem of nuclear safety cautiously and that, as information is developed, the precautions now taken may be relaxed and adequate safeguards still be maintained. The effectiveness of the new safety criteria at ORNL in reducing or eliminating activity releases will depend not only on how well the new standards are met but, as in the past, on the administrative control exercised. (H. N. Culver)

High-level Spill at the Hilac

On July 3, 1959, an incident occurred which resulted in the dispersion of curium with an activity level of about 10^{11} dis/min. This contamination was distributed throughout the Hilac (Heavy Ion Linear Accelerator) Building at the Lawrence Radiation Laboratory, University of California. A clear and comprehensive account of the incident and the subsequent decontamination operations have been reported by Garden and Dailey.[6]

Two curium targets, electroplated onto a 0.1-mil nickel foil covering a ¼-in.-diameter hole in a stainless-steel plate, were mounted in a chamber containing helium at atmospheric pressure. The beam from the accelerator was to pass through two similar nickel-foil windows in a degrading-foil assembly before striking the targets. The degrading-foil chamber separated the target chamber from the Hilac vacuum and, in effect, acted as a safety valve for the experiment since the window on the side facing the experimental assembly was always at a lower pressure differential than the window on the opposite side. On several occasions, when pressure was built up in the degrading-assembly chamber, the window away from the experiment blew out and released the pressure. At the time of the accident, the degrading chamber was being purged of air by sweeping helium through it. After a few minutes, the flow of purge gas was stopped, and a circulating pump was turned on. The experimenter failed, however, to open the return gas line; and, when the pressure reached about 9 psi, a window in the degrading-foil box blew out with explosive force. The blowout occurred this time in the window facing the target chamber. The rush of helium through the small opening apparently completely disintegrated both curium foils.

The experimental equipment was located inside a cave that was not airtight. The cave was built of concrete blocks with a wooden beam ceiling and was used as radiation protection from the accelerator beam. The flow of air through the cave was such that the contamination was quickly spread throughout the building. First notification that something was wrong was the sound of the explosion caused by the blowout of the foil window. The experimenter went into the cave, thinking that the window on the accelerator side had broken as in the past. An alpha

air monitor located in the cave area showed high activity, but unfortunately it was not fitted with an alarm system. The experimenter estimates that he spent 10 min in the high-activity area. The building was evacuated within 10 min after discovery of the release of activity by an announcement over a public-address system.

Twenty-seven people were in the building at the time of the accident. Five of these were exposed more than the others. None was judged to have been exposed to a large enough dose to warrant therapy. The largest exposure was received by the experimenter, and, although complete information is not yet available, it is estimated that he received less than the maximum permissible amount or, at the very most, only two or three times tolerance, based on plutonium data.

Decontamination of the building proceeded with utmost dispatch, seemingly by a well-conceived plan prearranged for such an event. Entry into the building was made by locating and taping down the contamination found in the walkway and then covering this area so that decontamination crews could work inward and outward from this clean area. This procedure of locating contamination and cleaning it or tying it down was followed throughout the decontamination work. The decontamination crews wore protective clothing and masks. Some equipment was cleaned in place, and some was covered and removed for decontamination elsewhere. It has been estimated that 90 per cent of the equipment will be cleaned and returned to use. Contamination was found everywhere. Overhead beams and hard-to-reach areas were carefully cleaned with vacuum cleaners, and the remaining contamination was taped down for later cleanup.

The total monetary loss attributed to the Hilac spill is $58,420, but this amount is quite insignificant when compared with lost time and potential damage to personnel. The building was out of use for three weeks for direct decontamination work and three more weeks for indirect cleanup.

It is the opinion of the reviewer that three major points were overlooked when setting up this experiment. First, the valving system for introducing helium into the degrading chamber should have been arranged with interlocks in such a way that it would not have been possible to pressurize the chamber; second, the experimental assembly should have been enclosed in its own airtight container; and, third, the alpha air monitors should have been fitted with an alarm.

The recommendations given by Garden and Dailey cover these points, as well as several others. They are as follows:[6]

1. A more clearly defined and forceful attitude regarding health and safety measures should be evidenced. The magnitude of contamination from this spill can probably be attributed, at least in part, to lack of firm policy.

2. Arrangements for obtaining the services of a physician quickly in case of a radiation incident should be made more workable. Although procedures have been set up for such an emergency, about an hour and a half elapsed before an M.D. could be reached for advice.

3. There should be more storage space for apparatus not being currently used. The presence of large amounts of extraneous equipment in the experimental areas greatly complicated the task of decontamination. Good housekeeping practices should be conscientiously enforced.

4. A shower with catch basin should be provided near an exterior door of the building for removing surface radioactive contamination from personnel involved in spills. The existing shower is designed for removal of chemicals and was in the area of greatest contamination. Hilac personnel found it necessary in this case to travel more than a quarter of a mile to a shower.

5. An automatic alarm system actuated by an alpha air monitor should be installed.

6. Primary enclosures (i.e., enclosures around individual pieces of experimental equipment containing radioactive materials) should be provided.

7. Secondary enclosures should be provided to isolate reasonable work areas. These areas should be individually ventilated.

(Oscar Sisman)

Calder Hall Operating Experience

The hazards involved in the operation of a nuclear facility for an extended period of time have been the subject of many reports of the type which are periodically reviewed in the various appropriate sections of *Nuclear Safety*. Most of the reports discuss the criteria upon which safety is based and analyze the types of accident which are possible, their consequences, and methods of designing to minimize the hazards both to the facility and to the general public.

The number of major incidents that have occurred at operating nuclear facilities has been

extremely small, and this has resulted from approaching the problem of safety in a conservative manner. From the few incidents that have occurred, much information concerning nuclear safety has been evolved. It must be recognized, however, that the experience gained during normal operation of a nuclear facility also introduces much information on which to base future designs and hazard studies. Since the safety or hazards of a reactor are closely coupled to the probability of malfunctioning of components in the system, the long-term operating experience of the world's reactors will do much to establish the degree of hazards involved in operation of nuclear facilities.

The operation of the Calder Hall reactors, for example, has resulted in the generation of a large amount of information upon which future designs may be based. In a recent lecture by Sir John Cockcroft,[7] the operating experience at the Calder Hall reactors was described, and significant features of the experience are discussed here.

Fuel-element Failures

The operation of the Calder Hall reactors through mid-July 1959 had resulted in the irradiation of about 90,000 fuel elements. The maximum burnup achieved was 1300 Mwd/ton, and the maximum time of exposure was about 19 months. Only 27 fuel-element failures had occurred, and they had resulted primarily from small leaks in the welds of the Magnox can or in the body of the can. These small leaks are believed to have developed because of cracks missed during inspection, handling damage, or strain during operation. It is reported that most of the failures occurred at low burnup, as would be anticipated if the failure were due to faulty inspection or handling. A large number of the 27 failures occurred in fuel elements, operating in the lower temperature regions of the core, and these failures are attributed to the strain developed in the Magnox cladding and the low ductility of Magnox at the lower temperatures. Since the elements that failed had not been subjected to high burnups, irradiation damage was not a cause of failure.

System Contamination

Since the failures of fuel elements might result in contamination of the equipment in the gas system, the heat exchangers and gas ducts at Calder Hall have been inspected. No appreciable contamination has been found to exist.

Wigner Energy

A large number of graphite specimens have been irradiated in the Calder Hall reactors for experimental measurements of stored energy. The experimental measurements give either the total stored energy or the rate of release of stored energy with increasing temperature (ds/dt). The results have indicated that the maximum value of the rate of release of stored energy with temperature is about 0.11 cal/(g)(°C), that the rate of accumulation of stored energy decreases rapidly with the temperature at which the graphite is irradiated, and that there is a tendency toward saturation with the irradiation dose. The ds/dt curves were extended to temperatures above 700°C, and it was found that ds/dt remains approximately constant above 350°C, with no evidence for any higher temperature peaks.

Reactor Kinetics

Kinetic studies performed on the Calder Hall reactors have provided information on both the fuel temperature coefficient and the moderator temperature coefficient. Although both coefficients are negative in the initial stages of operation, the accumulation of plutonium in the fuel causes the negative moderator coefficient to change to positive at burnups of about 400 Mwd/ton and to saturate at a positive temperature coefficient of 15×10^{-5} $(\delta k/k)/°C$. This positive coefficient causes any disturbances from an equilibrium power level to increase with a doubling time of about 2 min. The Calder Hall reactors are manually controlled; therefore the operator takes corrective action in a time short compared with the doubling time, and no difficulty is introduced. (H. N. Culver)

Action on Reactor Projects by Licensing and Regulating Bodies

In the course of securing final approval for the operation of a power reactor, the applicant must clear several hurdles designed to assure the AEC, and thus the public, that the safety of the public has been adequately considered in a responsible manner. The procedure involves

two major steps: (1) securing a construction permit and (2) securing an operating permit. Both these steps require the preparation of a hazards summary report, which must be reviewed and approved by both the Hazards Evaluation Branch (HEB) and the Advisory Committee on Reactor Safeguards (ACRS). As the final step before each license is issued, a public hearing is held. As a consequence of these various administrative reviews, any given reactor may be involved in reviews for a period of years. These reviews frequently result in modification of the reactor design or operation to enhance safety and, in any case, constitute the precedents for "adequate" safety.

Most power reactors proposed for construction and operation in this country are currently involved in some stage of the review process. The reactors discussed below have passed some milestone in the process during the quarter covered by this Review.

Pathfinder Plant of Northern States Power Company

The Pathfinder Atomic Power Plant is a 204-Mw(t) boiling-water unit with an internal superheater, which is a new developmental feature. One of the important objectives of operation of this plant will be the evaluation of the internal superheater. In addition to the superheater, an untried aluminum-clad uranium oxide fuel element, the controls necessitated by the unique fuel loading, and a new optional power-control system introduce special problems. The plant operator, Northern States Power Company, has research and design work in progress on these problems, which must be resolved before operation as proposed. Should these problems not be resolved by the time the plant is ready for startup, however, the plant is capable of operating in a conventional manner without these special features. Accordingly, the ACRS concluded at their December meeting that the reactor could be constructed at its proposed site "with reasonable assurance that it can be operated without undue hazard to the health and safety of the public." At the following ACRS meeting in January, Amendments 4 and 5 were reviewed and held not to change the Committee's opinion.

At a hearing on Feb. 15, 1960, to consider the issuance of a construction permit to the Northern States Power Company for the Pathfinder Plant, Beck of the HEB indicated his approval of the project but cited several novel features of the plant which require extensive research and development.[8] He included in this category: (1) the use of unproved aluminum-clad UO_2 fuel elements in the boiler region of the core, (2) the incorporation of a superheater as an integral part of the basic reactor core and the use of relatively unproved stainless-steel-clad stainless steel–UO_2 dispersion fuel elements in the superheater region, and (3) the use of a variable rate of primary coolant recirculation for controlling the power level of the core.

Beck indicated that clarification of the presently unresolved safety questions in the novel components proposed for this facility could be achieved through an appropriate research and development program. The substitution of Zircaloy-2 as an alternate cladding for the fuel in the boiler region would constitute a satisfactory alternate choice. Although the fuel proposed for use in the superheater section of the core has been used in other reactors, the conditions of its use in the Pathfinder are much more severe than in other reactors. If tests indicate that the fuel is not satisfactory for the conditions postulated for the Pathfinder, there is general agreement that the plant can be designed and operated safely as a straight boiling reactor without superheat. With respect to the novel method for controlling the power level of the core should recirculation control prove to be unacceptable, control rods could be used as a satisfactory alternate solution.

Later in the testimony by Beck it was indicated that the general containment concept was acceptable for the plant. However, in discussing the accident analyses of the applicant, Beck stated that the iodine exposures at the boundaries of the reactor "site as calculated by the applicant approached the range of hazard to the public which would not be acceptable" and that further study should be given to methods of reducing the potential exposure; methods suggested included reducing the leakage from the containment building or installing a spray system inside the building. In general, Beck indicated that additional information pertaining to the fuel performance for the conditions specified in the reactor was needed and that more analytical work on the consequences of fuel failure was required.

Examiner Jensch noted the proximity of a railroad to the reactor site (about 600 ft) and in-

quired if this was a precedent that the witness (Beck) would like to see repeated. Beck indicated that this was a general problem faced at many sites and that it was "a feature best not present but that it was accepted where prior arrangements have been made for traffic to be controlled in case of emergency."

Yankee Atomic Electric Company Power Plant

At its 23rd meeting, held Jan. 28 to 30, 1960, the ACRS considered the safety aspects of the Yankee Atomic Electric Company power plant. The reactor to be operated in this plant is a pressurized-light-water reactor[9] designed to operate at a thermal power level of 485 Mw. The ACRS had previously pointed out the novel features of the plant during its hearing pertaining to the granting of a construction permit.[10] At that time it was indicated that extensive investigations were required with respect to (1) the addition of soluble neutron absorbers, (2) the intention to permit nucleate boiling in the reactor, and (3) the large plutonium buildup. Following[11] the recent meeting, the ACRS stated:

> The problem of nucleate boiling and the use of boric acid as a soluble poison to supplement the control rods during cold shutdown have now been thoroughly investigated, and solutions to these problems satisfactory to the HEB and the Committee have been reported in the Final Hazards Summary Report.

In Amendment 7, the Yankee Atomic Electric Company made a proposal with respect to the determination of the effect of plutonium buildup[12] that was approved by the ACRS.[11]

On Jan. 13, 1960, Yankee submitted Amendment 18 to its license application. This amendment proposed to change the reactor design to allow continued operation of the plant in the event of leakage from the primary to the secondary system. In the amendment to the license application, Yankee states that:[13]

> These changes are based on design conditions which assume a leakage rate consistent with past experience on steam generators of tube and shell construction at a time when maximum expected activity may be present in the primary system. These changes will modify present design as follows:
>
> 1. Radiation monitoring will be provided in the discharge from the air ejector system.
>
> 2. Sufficient air dilution of the air ejector discharge will be provided to bring gaseous effluent activity concentration within acceptable limits.
>
> 3. The direct connection from the secondary blowdown system to the storm sewer system will be eliminated, and in its place a connection will be made from the blowdown system to the condenser circulating water discharge system, thereby providing sufficient dilution to bring liquid effluent activity concentration within acceptable limits.

Since design details pertaining to Amendment 18 were not established at the time of the 23rd meeting of the ACRS, the ACRS could not comment on the design; however, the ACRS did agree, in principle, that the changes proposed in the amendment would not create an undue hazard to the public. The ACRS concluded that the broad problems indicated at the time of the issuance of the construction permit had been resolved and that the reactor could be operated without undue risk to the health and safety of the public.

Vallecitos Boiling Water Reactor (VBWR) of the General Electric Company

At its 23rd meeting, the ACRS reviewed the final hazards summary report[14] for the VBWR,[15] including amendments up to and including Amendment 43. The Committee pointed out that the reactor control rods have two features that might cause difficulty, a pneumatic system for the rod motion and a sliding scale on the rod shafts.[16] However, the applicant intends to incorporate the following features in the design and operation to minimize difficulty with the control system. A separate pneumatic system will be used for each set of two rods. At least one pair of the rods will be tested initially at operating temperature and pressure a sufficient number of times to obtain reliable statistics on drop time. The remaining rods will be tested a sufficient number of times to show that all rods behave in a similar manner. Conventional lubricating oil will be eliminated from the pneumatic scram system, and the oil explosion hazards from the system will be minimized.

The Committee stated that:

> The worth of the control rods for each substantially different core should be measured and evaluated during core loading in terms of the excess reactivity possibly available in that core loading and in terms of any possible malfunction of the control conditions. This may limit or preclude certain types of operation.

In discussing the exclusion area at the reactor site which is presently being used for agricul-

tural purposes, the Committee recommended that "monitoring rules for people, crops, livestock and equipment be established." The Committee concluded that the plant could be modified as proposed and operated as discussed without undue hazard to the health and safety of the public.

Power Plant of the Nuclear Merchant Ship N.S. Savannah

The power plant of the nuclear merchant ship, the N.S. *Savannah*,[17] includes a pressurized-water reactor that generates 63.5 Mw(t) for the propulsion of a 22,000-ton vessel at 21 knots. (For additional information on the N.S. *Savannah*, see Safety in Nuclear Ship Propulsion, page 8 of this issue.) Although the N.S. *Savannah* is Commission-owned and therefore ACRS approval is not required by law, the vessel and its power plant have been reviewed by the Committee. Various aspects of design and operation at the Camden dock site had been considered at several previous meetings, and it was concluded, following the January ACRS meeting, that "in general, the design of the nuclear power system and its containment appears to be adequate for a nuclear propelled merchant ship, subject to proof of component integrity in an extensive test program." The Committee did, however, express concern regarding two aspects of the design which led to the recommendation "that a thorough study be made to resolve this question of upper limit [to the nuclear operation which may be carried out at Camden] and to analyze the problems of a specific alternate site." One of the items of concern was "an uncertainty as to the achievable filter efficiency of the iodine removal system." The other item of concern is embodied in the statement that "because of the prototype nature of the reactor during initial startup and early power operation, an unforeseen event might occur which would cause a major release of activity at a time when the containment system is inoperative."

Saxton Reactor

Authorization for the construction by the Saxton Nuclear Experimental Corporation of a 20-Mw(t) pressurized-water power plant was proposed by Jensch, AEC hearing examiner, in a recent intermediate decision.[18] Testimony at the hearing to consider the issuing of a construction permit indicated that the site was favorable from the safety standpoint and that the reactor presented no unusual problems which could not be solved by a satisfactory research and development program.[19] During the hearing it was pointed out that the Saxton reactor is substantially similar to both the Yankee reactor and the Belgian reactor BR-3 and that both of these reactors are scheduled to be completed and operated for some time prior to the startup of the Saxton reactor.

Carolinas-Virginia Tube Reactor (CVTR)

The ACRS has at various meetings considered the safety aspects of the CVTR, and in a letter dated Sept. 14, 1959,[20] the ACRS pointed out to the Commission that the 60-Mw(t) CVTR contains "a number of features which have not been used in previous power reactors" and that some of these features "have not been completely developed." Some of the problems were pointed out in the letter, which went on to say that "the applicant has not yet proposed alternate solutions using presently known technology which may be substituted in case his research and development program does not confirm his expectations." Therefore the ACRS could not conclude at that time that the reactor could be constructed and operated at the proposed site without undue hazard to the health and safety of the public.

The ACRS, at its 22nd meeting, again considered the CVTR for a construction permit.[21] At this meeting, Amendment 4 to the preliminary hazards summary report was presented, as well as oral information, to supply additional information pertaining to the suitability of Zircaloy for pressure tubes and the determination of values of nuclear, thermal, and hydrodynamic design parameters. In its letter of Dec. 14, 1959, the Committee concluded "that a reactor of this general type with containment as proposed by the Associates for the Parr location may be constructed with reasonable assurance that it can be operated at the site selected without undue risk to the health and safety of the public."

A hearing was then held on Feb. 23, 1960, to consider the issuance of a construction permit for the CVTR.[22,23] In testimony at the hearing, several features of the reactor were discussed by Beck. In his testimony it was pointed out that

a large portion of the well-known engineering technology of pressurized-water reactors is applicable to the CVTR. He indicated that the new problems arising from the use of heavy water, separate moderator and coolant systems, and the U-shaped pressure tubes have yet to be worked out but that the research and development program planned by the applicant, as well as information from similar reactors which will operate earlier than the CVTR, will be adequate to resolve the present reservations with respect to the safety of the proposed reactor. Specifically mentioned were the use of Zircaloy as the structural material for the pressure tubes, particularly from the standpoint of corrosion resistance, and the life expectancy of the tubes under neutron irradiation. In addition, Beck pointed out the conceptual design problems which have yet to be worked out for the use of heavy water and the U-shaped Zircaloy pressure tubes. In discussing the accident analysis, Beck indicated that the applicant's survey of the loss-of-coolant-flow accident is not sufficiently complete to assure that no fuel-element damage would result from a sudden loss of coolant flow to the core.

During the course of the hearing, the hearing examiner brought up the question of whether conclusions made with respect to the prototype CVTR could meet an issue specified by AEC for reactors of this type in general. Beck stated that "it was not necessary to define reactor type in terms of previous types to make a safety determination, and that an operating license could be issued without there being a precedent for the reactor type." He further stated that "it was difficult to put this reactor into a class of reactors already operated, and that there has been no precedent for this reactor in type."

(H. N. Culver and W. B. Cottrell)

Safeguards Reports and Selected Reading

The recently issued safeguards reports and other selected literature pertaining to hazards and safeguards of reactors are listed below for reference. Because of the similarity of many reactors (in particular, the research reactors), the list is not intended to be all-inclusive. In addition to the hazards reports listed below, hazards reports which provide the bases for the several articles which compare various features of U. S. power reactors are listed at the end of the appropriate section.

1. Hazards Summary Report, Consolidated Edison Thorium Reactor, Rev. 1, Exhibit K-5, January 1960.

2. General Atomic Division, General Dynamics Corp., Safeguards Report for the MGCR Critical Experiment, Report GA-1100, Dec. 18, 1959.

3. Philadelphia Electric Co., Site Evaluation Report for Peach Bottom Atomic Power Station, Parts 1 and 2, December 1959.

4. G. S. Brunson, Design and Hazards Report for the Argonne Fast Source Reactor (AFSR), USAEC Report ANL-6024, Argonne National Laboratory, June 1959.

5. Aerojet-General Nucleonics, Army Gas-Cooled Reactors System Program: Preliminary Hazards Summary Report for the ML-1 Nuclear Power Plant, USAEC Report IDO-28537, Apr. 22, 1959.

6. Puerto Rico Water Resources Authority and General Nuclear Engineering Corp., Boiling Nuclear Superheater (BONUS) Power Station: Preliminary Design Study and Hazards Summary Report, USAEC Report TID-8524, four volumes, to be published.

7. National Aeronautics and Space Administration, Final Hazards Summary for NASA Plum Brook Reactor Facility, Parts 1, 2, and 3, December 1959.

8. N. G. Wittenbrock, P. C. Walkup, and J. K. Anderson, eds., Plutonium Recycle Test Reactor — Final Safeguards Analysis, USAEC Report HW-61236, Hanford Atomic Products Operation, Oct. 1, 1959.

9. W. A. Vogele et al., Summary Report of Reactor Hazards Associated with the High Temperature Test Facility, USAEC Report WAPD-SC-630, Westinghouse Electric Corp., Bettis Atomic Power Div., August 1959.

10. E. A. Wimunc and J. M. Harrer, Hazards Evaluation Report Associated with the Operation of EBWR at 100 Mw, USAEC Report ANL-5781 (Addendum), Argonne National Laboratory, December 1959.

11. Pratt and Whitney Aircraft Div. of United Aircraft Corp., Hazards Summary Report on Critical Experiment Program No. 2 at CANEL—October 1958, USAEC Report PWAC-283, Dec. 15, 1959. (Classified)

12. R. O. Williams, Jr., ed., Hazards Analysis of the Organic Moderated Reactor Experi-

ment, USAEC Report NAA-SR-4065, Atomics International, Dec. 15, 1959.

13. E. O. Swickard, comp., Los Alamos Molten Plutonium Reactor Experiment (LAMPRE) Hazard Report, USAEC Report LA-2327, Los Alamos Scientific Laboratory, June 1959.

14. H. Blatz, ed. in chief, *Radiation Hygiene Handbook*, McGraw-Hill Book Company, Inc., New York, 1959.

15. R. J. Smith, Reactor Safety: A Literature Search, USAEC Report TID-3525(Rev. 1), November 1959.

16. J. C. Mailen, Reactor Coolant Decontamination: A Literature Survey, USAEC Report CF-59-10-124, Oak Ridge National Laboratory, Oct. 30, 1959.

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1. Plutonium Release from the Thorex Pilot Plant, *Nuclear Safety*, 1(3): 78–80 (March 1960).
2. F. N. Browder, comp. and ed., Radioactive Waste Management at Oak Ridge National Laboratory, USAEC Report ORNL-2601, Apr. 17, 1959.
3. Oak Ridge National Laboratory, based on written and oral information from the Chemical Technology, Health Physics, and Operations Divisions as compiled by the Radioactive Operations Review Committee.
4. C. J. Borkowski et al., Committee's Report on Ruthenium Fallout Incident, USAEC Report ORNL CF-60-1-116, Oak Ridge National Laboratory, Jan. 25, 1960.
5. Letter from J. A. Swartout to Division Directors and Chairman of Safety Review Committee regarding General Criteria for Containment of Radioactive Operations, Dec. 11, 1959.
6. N. B. Garden and C. Dailey, High-level Spill at the Hilac, USAEC Report UCRL-8919, University of California Lawrence Radiation Laboratory, Sept. 24, 1959.
7. J. Cockcroft, Calder Operating Experience, Monthly Information Bulletin of United Kingdom Atomic Energy Authority, *Atom*, 37: 14–20 (November 1959).
8. *Atomic Energy Clearing House*, 6(8): 7–12 (Feb. 22, 1960).
9. Yankee Atomic Electric Co., Nuclear Power Station Technical Information and Final Hazards Summary Report. Part B, License Application, AEC Docket No. 50-29, USAEC Report YAEC-167 (Vols. I and II), 1959.
10. Letter from the Advisory Committee on Reactor Safeguards to the Chairman, U. S. Atomic Energy Commission, Sept. 16, 1957.
11. Letter from L. Silverman to Hon. John A. McCone, February 1960, subject: Yankee Nuclear Power Station of the Yankee Atomic Electric Co.
12. *Nuclear Safety*, 1(1): 43 (September 1959).
13. Amendment No. 18 to License Application, dated June 29, 1959, AEC Docket No. 50-29 (see reference 9 above).
14. General Electric Co., Vallecitos Atomic Laboratory, General Electric Vallecitos Boiling Water Reactor Final Hazards Summary Report, Report SG-VAL-2(3rd Ed.), Nov. 30, 1959.
15. Amendment No. 43 to License Application for Vallecitos Boiling Water Reactor, Jan. 22, 1960.
16. Letter from L. Silverman to Hon. John A. McCone, Feb. 8, 1960, subject: General Electric Vallecitos Boiling Water Reactor.
17. J. H. MacMillan, Babcock and Wilcox Co., Sept. 15 and Nov. 3, 1958. (Unpublished)
18. *Atomic Energy Clearing House*, 6(4): 13–14 (Jan. 25, 1960).
19. *Atomic Energy Clearing House*, 5(51): 11–16 (Dec. 21, 1959).
20. *Atomic Energy Clearing House*, 5(40): 3–4 (Oct. 5, 1959).
21. *Atomic Energy Clearing House*, 6(2): 6 (Jan. 11, 1960).
22. *Atomic Energy Clearing House*, 6(4): 17–18 (Jan. 25, 1960).
23. *Atomic Energy Clearing House*, 6(9): 8–15 (Feb. 29, 1960).

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U.S. GOVERNMENT PRINTING OFFICE: 1960—522537/4